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ORIGINAL

Review and Evaluation of the Millstone Unit 3 Probabilistic Safety Study

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Containment Failure Modes, Radiological Source-Terms and Offsite Consequences

ORIGINAL

Prepared by M. Khatib-Rahbar, W. Pratt, H. Ludewig/BNL
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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

A technical review and evaluation of the Millstone Unit 3 probabilistic safety study has been performed. It was determined that; (1) long-term damage indices (latent fatalities, person-rem, etc.) are dominated by late failure of the containment, (2) short-term damage indices (early fatalities, etc.) are dominated by bypass sequences for internally initiated events, while severe seismic sequences can also contribute significantly to early damage indices. These overall estimates of severe accident risk are extremely low compared with other societal sources of risk. Furthermore, the risks for Millstone-3 are comparable to risks from other nuclear plants at high population sites. Seismically induced accidents dominate the severe accident risks at Millstone-3. Potential mitigative features were shown not to be cost-effective for internal events. Value-impact analyses for seismic events showed that a manually actuated containment spray system might be cost-effective.

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1. INTRODUCTION

1.1 Background

After the accident at Three Mile Island, the United States Nuclear Regulatory Commission (USNRC) recognized the need to reexamine the capabilities of nuclear power plants to accommodate the effects of hypothetical severe accidents beyond the design basis. This reexamination included consideration of potential design modifications to mitigate the consequences of these degraded and core melt accidents.

The Zion and Indian Point (Z/IP) power plants were chosen to initiate this activity because the populations surrounding the two sites are larger than other reactor sites. The concern was that due to the proximity of these two sites to high population densities, they could comprise a disproportionately high component of the total societal risk from U.S. commercial nuclear power programs.

As part of this continuing effort, programs have been undertaken to evaluate the risk from other plant sites situated near high population centers in order to examine the need for design modifications or mitigation features which could potentially reduce the accident risks at these facilities. Millstone-3 is one of these plants.

Probabilistic Risk Assessment (PRA) studies have been undertaken by a number of utilities[1-3] and reviewed by Brookhaven National Laboratory (BNL) under contract to the USNRC. The staff's initial contribution to the understanding of severe accident progression and mitigation specifically for the Z/IP facilities is presented in NUREG-0850.[4]

In August 1983, Northeast Utilities Inc. completed a probabilistic safety study for Millstone-3. The Millstone-3 plant, which is currently undergoing operating license review, is located near a relatively high population center in southeastern Connecticut. The Millstone-3 Probabilistic Safety Study (MPSS-3)[5] included a probabilistic evaluation of core melt frequencies, an analysis of containment failure modes and radionuclide releases, and an assessment of radiological consequences.

This report describes the review and evaluation by the USNRC staff and their contractors at BNL of the containment failure modes, radiological source terms and offsite consequences for the MPSS-3.

1.2 Objectives and Scope

The objective of this report is to provide a severe accident risk perspective as a basis for examining the need for potential safety improvements to the Millstone-3 plant. The report describes severe accident phenomenology, containment response, radionuclide releases, and offsite consequences. Core melt frequencies are discussed in a separate report which was prepared for the Reliability and Risk Assessment Branch of NRC by the Lawrence Livermore National Laboratory (LLNL).[6]

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In the present report, principal containment design features are discussed and compared with those of Indian Point and Zion. Those portions of the MPSS-3 related to severe accident phenomena, containment response, source terms and consequences, are described and evaluated. Numerical adjustments to the MPSS-3 estimates are documented and justified.

Severe accident risk estimates are presented and some uncertainty estimates are described. The contributions to risk from individual accident sequences and containment failure modes are presented. An assessment of potential severe accident mitigative features is also presented.

1.3 Organization of the Report

A brief review of the Millstone-3 plant design and features is presented in Chapter 2 along with comparisons to Zion and Indian Point plant designs. Chapter 3 contains the assessment of containment performance. Specifically, analytical methods, containment event trees, accident phenomenology, the containment matrix, and radiological source terms are reviewed. Where adjustments to the applicant's calculations were deemed necessary, changes have been documented and justified. Chapter 4 addresses the approach used by the NRC staff to calculate the offsite consequences. In Chapter 5, the NRC/BNL severe accident risk estimates are presented and discussed; Chapter 6 includes a discussion of uncertainties. Cost-benefit analyses to evaluate potential mitigative design improvements are described in Chapter 7. The results of this review are summarized in Chapter 8.

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2. PLANT DESIGN FEATURES IMPORTANT TO SEVERE ACCIDENT ANALYSIS

In this section, those plant design features that may be important to an assessment of degraded core and containment analysis are reviewed. These important features are then compared with the Zion and Indian Point facilities in order to identify commonalities and differences for benchmark comparisons.

2.1 Assessment of Plant Design

Millstone-3 is a four-loop Pressurized Water Reactor (PWR). The Nuclear Steam Supply System was designed by Westinghouse; the major balance of plant systems and the containment were designed by Stone and Webster.[5]

The plant is a 3411 MWt (1150 MWe) power reactor employing the Westinghouse 17 x 17 core design. The reactor coolant system is a four-loop configuration with U-tube recirculating steam generators. The emergency core cooling system consists of four accumulators containing 7100 gallons of water each, which are designed to discharge when the reactor coolant system pressure falls below 600 psia, and a safety injection system which draws water from a 1.2 million gallon refueling water storage tank and delivers it to the reactor coolant system via either the charging pumps, high head safety injection pumps or low head safety injection pumps. The long-term core cooling is attained by a completely independent recirculation cooling system (whose major components are shared with the recirculation spray system) which consists of four pumps and four heat exchangers which are cooled by the service water system.

The auxiliary feedwater system also provides a core cooling function by removing heat from the RCS after reactor shutdown via the steam generators. This system, which consists of two motor driven pumps and one turbine driven pump takes suction from the condensate storage tank.

The Millstone containment structure is a carbon steel-lined, reinforced concrete structure with a net free volume of about 2,260,000 cubic feet. The Millstone containment design uses the subatmospheric containment concept. During normal operation, the containment atmosphere will be maintained at a subatmospheric pressure of approximately 9 to 12 psia.

The secondary containment in the Millstone design is comprised of the containment enclosure building and the associated supplementary leak collection and release system (SLCRS) provided to mitigate the radiological consequences of postulated design basis accidents.

The Millstone design includes two independent active containment heat removal systems (CHRS) (Figure 2.1). These are the quench spray system (QSS) and the recirculation spray system (RSS). The containment air coolers are not considered part of the CHRS. The CHRS is designed to depressurize the containment to a subatmospheric condition within one hour following a high energy line break accident.

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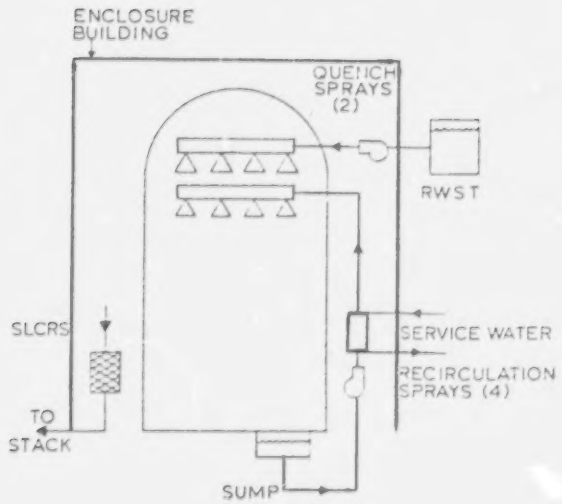

Figure 2.1 Schematic of the Millstone-3 Containment Cooling System

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The QSS consists of two redundant 100 percent-capacity trains, each containing a quench spray pump, a chemical injection system and riser pipes leading to two common 360° quench spray headers. Rated flow to the quench spray headers is approximately 4000 gpm with one quench spray pump operable and 6000 gpm with both pumps operable. Each redundant quench spray subsystem draws water independently from the 1.2 million gallon refueling water storage tank. The QSS is actuated automatically upon receipt of the containment depressurization signal (CDA). The CDA signal is initiated by high containment pressure (24.7 psig). The quench spray is terminated when the RWST reaches a predetermined low level.

The RSS consists of two parallel redundant 100-percent-capacity trains, each containing two containment recirculation pumps with dedicated heat exchangers, and riser pipes leading to two common 360° recirculation spray headers. The rated flow for each recirculation pump is about 3900 gpm. The four redundant recirculation spray subsystems take suction from the containment sump; the recirculation spray water flows through recirculation coolers where it is cooled by the service water. The RSS pumps are started automatically approximately 4 minutes after receipt of the CDA signal.

An important factor in the reliability of recirculation spray for PWRs is the susceptibility of the containment sump to blockage. The principal threat of sump blockage is the insulation material which would be scattered by the jet created in large or intermediate break loss of coolant accidents (LOCA's). The ability of the sump to remain operational is a confirmatory item in the licensing review of Millstone-3. The significance of this issue lies not only in its effect on spray operation, but also on the likelihood that large and intermediate break LOCA's will result in core melt as a result of pump failure due to loss of NPSH. Because of the large debris screen area in the Millstone-3 sump, and because of the favorably low heat-loss characteristics of the insulation on the Millstone-3 primary coolant system, the staff has concluded that this effect will not contribute significantly to risk.

For certain severe accidents, core debris may be dispersed from the reactor cavity. The potential for this debris to be transported to the containment sump and to impact the performance of the recirculation spray pumps (via either sump screen blockage or debris ingestion) has not been addressed in the PRA. The likelihood of transporting a significant amount of core debris to the sump area is considered to be small for two reasons. First, the configuration of the reactor cavity is such that the bulk of corium released from the reactor vessel would be trapped within the cavity, with only the smaller particles capable of exiting the cavity. Second, the region of containment into which the exiting debris would enter is not swept by the containment sprays; thus the potential for transport of debris to the sump is reduced.

The hydrogen recombiner system consists of two redundant thermal-type hydrogen recombiners and associated control units located in the recombiner building. Each recombiner train has a capacity of 50 SCFM and is designed to seismic category I design criteria. The recombiner system is supplied from the Class IE emergency buses, and is manually started and operated from a

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local control panel. Operation of the hydrogen recombiner system during a severe accident would not significantly impact the accident progression since the predicted rates and magnitude of hydrogen generation and release for such accidents far exceed the capacity of the recombiner system.

The containment geometry in the area underneath and around the reactor vessel precludes water from entering the reactor cavity area until a major portion of the Refueling Water Storage Tank (RWST) has been exhausted via the quench spray system (see Figure 2.2). The containment design also includes a permanent seal ring between the reactor vessel flange and the biological shield walls, which would prevent introduction of water into the reactor cavity from either break flow or spray flow in the area of the reactor vessel or the refueling cavity. This is referred to as a dry cavity configuration. The cavity geometry is expected to suppress the dispersion of core debris from the reactor cavity to the general containment area following failure of the reactor vessel during core melt sequences. The cavity area geometry also would reduce the potential for establishing effective convective air currents between the cavity and general containment area for heat removal from core debris in the reactor cavity area.

The containment building basemat and the internal concrete structures are composed of basaltic-based concrete. As concrete is heated, water vapor and other gases are released. The initial gas release consists largely of carbon monoxide, carbon dioxide, the quantity of which depends on the amount of calcium carbonate in the concrete mix. Limestone concrete can contain up to 80% calcium carbonate by weight, which could yield up to 53 lb of carbon dioxide per cubic foot of concrete. However, basaltic-based concrete contains very little calcium carbonate and would not therefore release a significant amount of carbon dioxide.[4]

2.2 Comparison to Zion and Indian Point Plant Designs

Table 2.1 sets forth the design characteristics of the Zion (Units 1 or 2) and the Indian Point (Unit 2) facilities as they compare to the Millstone Unit 3 plant.

It is seen that the three plants are quite similar in containment building and primary system design although there are differences in the containment cooling mechanisms, the lower reactor cavity configuration, RWST volume and the chemical compositions of the concrete mix.

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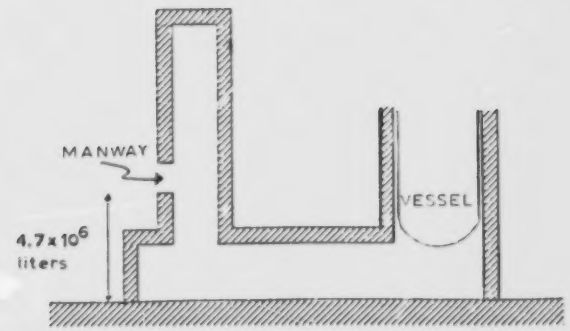

Figure 2.2 Schematic of the Millstone-3 Lower Cavity Configuration

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Table 2.1 Comparison of Design Characteristics

Design Parameters		Zion Unit 1[1,4]	Indian Point Unit 2[3,4]	Millstone Unit 3[5]
Reactor Power	[MW(t)]	3250	3030	3411
Containment Building:				
Free Volume	(ft^3)	2.73×10^6	2.61×10^6	2.26×10^6
Design Pressure	(psia)	62	62	59.7
Initial Pressure	(psia)	15	14.7	12.3/8.9*
Initial Temperature	(°F)	120	120	120/80
Primary System:				
Water Volume	(ft^3)	12,710	11,347	11,695*
Steam Volume	(ft^3)	720	720	380*
Mass of UO_2 in Core	(lb)	216600	216600	222739
Mass of Steel in Core	(lb)	21,000	20,407	
Mass of Zr in Core	(lb)	44,500	44,600	45,296
Mass of Bottom Head	(lb)	87,000	78,130	87,000
Bottom Head Diameter	(ft)	14.4	14.7	14.4
Bottom Head Thickness	(ft)	0.45	0.44	0.45
Containment Building Heat Removal:				
Sprays		yes	yes	yes
Fans		yes	yes	no
Accumulator Tanks:				
Total Mass of Water	(lb)	200,000	173,000	236,000*
Initial Pressure	(psia)	665	665	665*
Temperature	(°F)	150	150	80
Refueling Water Storage Tank:				
Total Mass of Water	(lb)	2.89×10^6	2.89×10^6	1×10^7
Initial Pressure	(psia)	14.7	14.7	14.7
Temperature	(°F)	100	120	50/40
Reactor Cavity:				
Design		Wet	Wet	Dry
Concrete Material		Limestone	Basaltic	Basaltic

*These values were taken from the Millstone-3 FSAR and Technical Specifications.

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3. CONTAINMENT RESPONSE AND RADIOLOGICAL SOURCE TERMS

In this chapter, the review of containment response to severe accidents is described. The timing and mode of containment failure, as well as the radiological source terms are examined and discussed.

3.1 Description of Plant Damage States and Containment Response Classes

In the MPSS-3, each core melt accident sequence is assigned to one of the plant damage states listed in Table 3.1. Summation over all of the frequencies of core melt accidents associated with a given plant damage state yields the annual frequency of the damage state.

The plant damage states classify events according to three parameters;

(1) Initiating Event, namely:

- A - Large break Loss-of-Coolant Accident (LOCA)
- S - Small break LOCA
- S' - Incore instrument tube LOCA
- T - Transient
- V2 - Steam Generator Tube Rupture (SGTR)
- V3 - Seismic induced large LOCA with failure of ECC injection combined with containment bypass
- V - Interfacing systems LOCA

(2) Timing of Core Melt, namely:

- E - Failure of Emergency Core Cooling Injection (ECCI)
- L - Failure of ECC recirculation

(3) Status of Containment Heat Removal (CHR), namely:

- $\overline{}$ - Complete loss of Containment Sprays (CS)
- C' - Loss of recirculation CS
- C" - Loss of quench CS
- C - All spray systems available

From the viewpoint of containment response, many of the plant damage states can be grouped into containment classes. The classes defined in Table 3.2 are differentiated primarily according to spray availability. The frequency of each containment class is the sum of the frequencies of the plant damage states assigned to the class.

Annual plant damage state frequencies calculated by the applicant for both internal and external events were reviewed and evaluated by the Reliability and Risk Assessment Branch (RRAB) of the U.S. Nuclear Regulatory Commission (NRC) with assistance from the Lawrence Livermore National Laboratory (LLNL).[6] Table 3.3 presents the containment class frequency estimates of RRAB for internal events and fires. Also included are seismic containment

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Table 3.1 Notation and Definitions for Plant Damage States

Symbol	Description
AEC	Large LOCA, Early Melt
AEC'	Large LOCA, Early Melt, Failure of Recirculation Spray
AE	Large LOCA, Early Melt, No Containment Cooling
ALC	Large LOCA, Late Melt
ALC'	Large LOCA, Late Melt, Failure of Recirculation Spray
ALC"	Large LOCA, Late Melt, Failure of Quench Spray
AL	Large LOCA, Late Melt, No Containment Cooling
SEC	Small LOCA, Early Melt
SEC'	Small LOCA, Early Melt, Failure of Recirculation Spray
SE	Small LOCA, Early Melt, No Containment Cooling
S'E	Incore Instrument Tube LOCA, Early Melt, No Containment Cooling
S'EC	Incore Instrument Tube LOCA, Early Melt
SLC	Small LOCA, Late Melt
SLC'	Small LOCA, Late Melt, Failure of Recirculation Spray
SLC"	Small LOCA, Late Melt, Failure of Quench Spray
SL	Small LOCA, Late Melt, No Containment Cooling
S'L	Incore Instrument Tube LOCA, Late Melt, No Containment Cooling
TEC	Transient, Early Melt
TEC'	Transient, Early Melt, Failure of Recirculation Spray
TE	Transient, Early Melt, No Containment Cooling
V2EC	Steam Generator Tube Rupture, Steam Leak, Early Melt
V2EC'	SGTR, Early Melt, Failure of Recirculation Spray
V2E	SGTR, Early Melt, No Containment Cooling
V2LC	SGTR, Late Melt
V2LC'	SGTR, Late Melt, Failure of Recirculation Spray
V2LC"	SGTR, Late Melt, Failure of Quench Spray
V2L	SGTR, Late Melt, No Containment Cooling
V	Interfacing Systems LOCA
V3	Seismically Induced AE Combined with Containment Bypass

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Table 3.2 Containment Response Class Definitions

Containment Class	Plant Damage States
1	AE
2	SE
3	AL
4	TE
5	SL
6	AEC, ALC, SEC, SLC, TEC, S'EC
7	TEC', SLC'
8	AEC', ALC', SEC'
9	ALC", SLC"
10	S'E, S'L
	V
	V2EC, V2EC', V2E, V2LC, V2LC', V2LC", V2L
	V3

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Table 3.3 Containment Class Mean Frequencies (Per Reactor Year)*

Containment Class	Plant Damage States	Internal Events	Fires	Seismic Dames & Moore	Seismic SHCP
1	AE	-	-	5E-7	1E-5
2	SE	1.3E-5	-	6E-6	1E-4
3	AL		-	-	-
4	TE	9E-7	3E-6	-	1E-6
5	SL		-		
6	AEC, ALC, SEC, SLC, TEC, S'EC	1.0E-4	3E-6	-	-
7	TEC', SLC'	6E-7	2E-7	-	-
8	AEC', ALC', SEC'	3E-7	1E-7	-	-
9	ALC", SLC"	-	-	-	-
10	S'E, S'L	2E-7	-	-	-
	V	8E-7	-	-	-
	V2EC, V2EC', V2E, V2LC, V2LC', V2LC" V2L	2.5E-6	-	-	-
	V_3	-	-	1E-7	4E-6

*Frequencies are based on the review of Millstone-3 PSS by the NRC staff, with assistance from the Lawrence Livermore National Laboratory.[6] Frequencies less than 1E-7 per reactor year have been neglected.

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class frequency estimates based on two seismic hazard curves, one developed by the applicant's contractor (Dames and Moore), and one developed by the Lawrence Livermore National Laboratory as part of the Seismic Hazard Characterization Project (SHCP).[7] The staff believes that the severe accident risk estimates attributable to seismic events are bounded on the high end by the SHCP results and on the low end by the Dames and Moore results.

In order to comprehensively assess the risk from seismic events, it is necessary to make separate consequence calculations for those accidents which are initiated by earthquakes severe enough to impair evacuation. For this purpose, both the Dames and Moore and SHCP seismic frequency estimates are divided into two categories (Table 3.4). The seismic events with peak ground acceleration below 0.5 g can be binned with internal events and fires. Seismic events with acceleration greater than 0.5 g are judged to impair evacuation, and are treated separately in the consequence analysis (Table 3.5). More details of the consequence analysis are given in Chapter 4.

3.2 Containment Analysis Methods

A brief description of the computer codes used in the MPSS-3 to perform the transient degraded core and containment response analyses is provided in this section.

Table 3.6 summarizes the code package as applied to various phases of the accident. It is seen that the MARCH code is used to model the core and primary system behavior and to obtain the steam and water energy releases for (1) the entire transient in the case of non-dispersal accident events and (2) until the vessel failure for the dispersal (high pressure) scenarios. These mass and energy releases form the input for the other computer codes used to evaluate the containment response for the non-dispersal cases (Figure 3.1).

For sequence classes in which the reactor coolant system remains at an elevated pressure until the vessel failure (dispersal cases), the MODMESH code is used. This code calculates the steam and hydrogen blowdown from the reactor vessel using an isothermal ideal gas model. The water boil-off from the reactor cavity floor is modeled using a saturated critical heat flux correlation. Additionally, the accumulator discharge following primary system depressurization caused by the vessel failure is also considered.

For the non-coolable debris bed and core-concrete interaction, the INTER subroutine of MARCH is replaced by the CORCON-MOD1 code modified by Westinghouse. The output from MARCH or CORCON is used as input, after preprocessing by MODMESH, to the COCOCLASS9 code. The COCOCLASS9 code replaces the MACE subroutine of the MARCH code. In the COCOCLASS9 code, the containment steam/water, noncondensibles, and the sump water are modeled by a single volume. The code also includes a structural heat transfer model, hydrogen combustion, and capability for containment heat removal through containment sprays and sump recirculation spray systems, as described in Section 4.3.2 and Appendix 4-E of the MPSS-3 report.[5]

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Table 3.4 Breakdown of Seismic Frequencies by Degree of Severity (Per Reactor Year)

Containment Class	Frequency (Per Reactor Year)			
	Dames & Moore		SHCP	
	<.5g	>.5g	<.5g	>.5g
1	7.8E-8	4.2E-7	1.1E-6	8.9E-6
2	3.1E-6	2.9E-6	4.0E-5	6.0E-5
4	-	-	3.5E-7	6.5E-7
V_3	1.6E-8	8.4E-8	3.0E-7	3.7E-6

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Table 3.15 BNL/NRC Release Category Summary (Internal and External Events)

Category	Release Duration (hrs)	Release Energy (Btu/hr)	Fission Product Release Fractions							
			Xe-Kr	OI	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
M-1A	1.0	0.5 E6	1E-0	7E-3	4.8E-1	7.9E-1	4.4E-1	9E-2	4E-2	6E-3
M-1B	1.0	0.5 E6	9E-1	7E-3	7E-2	5E-2	3E-2	6E-3	2E-3	4E-4
M-2A	2.0	150 E6	7E-1	5E-3	5E-1	6E-1	2E-1	7E-2	2E-2	3E-3
M-2B	0.5	520 E6	9E-1	-	7E-1	4E-1	4E-1	5E-2	4E-1	3E-3
M-3	2.0	190 E6	8E-1	5E-3	5E-1	6E-1	2E-1	8E-2	3E-2	3E-3
M-4	2.0	70 E6	9E-1	6E-3	2E-1	6E-1	5E-1	7E-2	5E-2	7E-3
M-5	0.5	150 E6	9E-1	6E-3	1E-2	5E-1	5E-1	5E-2	4E-2	6E-3
M-6	0.5	150 E6	9E-1	6E-3	1E-2	5E-1	5E-1	5E-2	4E-2	7E-3
M-6S	0.5	70 E6	9E-1	-	7E-3	1E-3	1E-3	1E-4	9E-5	1E-5
M-7	0.5	150 E6	9E-1	6E-3	9E-3	3E-1	3E-1	3E-2	2E-2	4E-3
M-8	0.5	22 E6	9E-1	7E-3	8E-3	1E-5	1E-5	1E-6	1E-6	2E-7
M-9	0.5	22 E6	9E-1	6E-3	2E-3	2E-6	1E-6	2E-7	9E-8	1E-8
M-10	10.0	n/a	3E-1	2E-3	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
M-11	10.0	n/a	6E-3	2E-5	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7
M-12	5.0	n/a	1E-3	9E-6	6E-6	1E-6	9E-7	2E-7	8E-8	1E-8

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Table 3.6 Summary of MPSS-3 Computational Tools

Computer Code	Accident Phase
MARCH	1. Non-dispersal Events - Total Transient 2. Dispersal Events - Initial blowdown, slump, and vessel failure
MODMESH	1. Non-dispersal Events - Interface to other codes 2. Dispersal Events - Discharge and scatter within the reactor cavity, cavity boil-off
CORCON-MOD1/W	Core-concrete interaction for dry cavity
COCOCLASS9	Containment building pressurization and hydrogen combustion
CORRAL-II	Fission product transport in containment
CRAC2	Consequences

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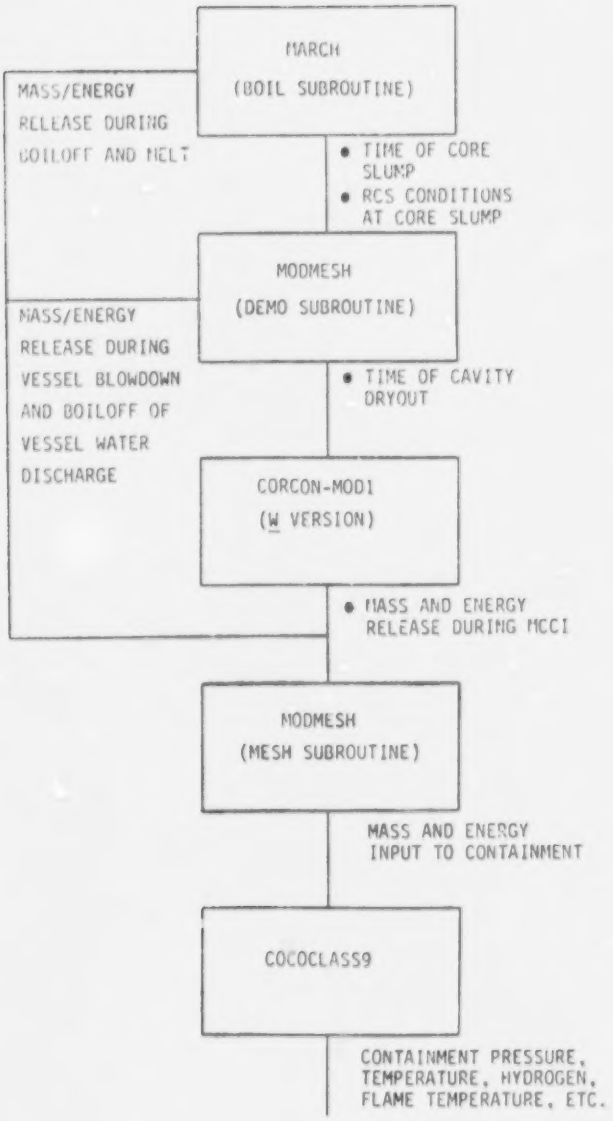

Figure 3.1 The MPSS-3 Computational Approach

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Fission product transport and consequence calculations are performed using the CORRAL-II and CRAC-2 computer codes, respectively.

Benchmark studies of the containment response codes are performed in Appendix 4-I of the MPSS-3. There, it is shown that, for a simulated "design basis" large break loss-of-coolant accident, COCOCLASS9 results are in agreement with the results predicted by (1) the LOCTIC containment response code, used for licensing calculations, and (2) the MACE subroutine of MARCH1.1. It must be noted that these analyses were performed only for the case without any core degradation, and therefore the containment atmosphere was close to saturation.

Due to the limited scope of the present review, detailed audits of calculational results were not made; however, the results of Containment Loads Working Group (CLWG) studies for the subatmospheric containment[8] and other generic studies were used to verify the validity of the MPSS-3 estimates. Several important changes to the containment response matrix and source term estimates were made.

The containment failure pressure probability distribution (Fig. 3.2) was not evaluated as part of this review.

3.3 Containment Event Tree and Accident Phenomenology

An important step towards the development of the containment matrix involves the quantification of branch point probabilities in the containment event trees. These probabilities depend heavily on the analyses of degraded core phenomenology and the containment building response described in Sections 4.2 through 4.7 of the MPSS-3.[5]

In the MPSS-3, the containment event tree is divided into six distinct time frames, which represent the time phases during an accident event in which potential containment failure is considered. Table 3.7 is reproduced from MPSS-3 and summarizes the six time frames along with the corresponding containment event tree nodal questions.

3.4 Containment Matrix (C-Matrix)

The sixteen nodes in the Millstone-3 containment event trees are outlined in Table 3.7. A negative response at any of seven nodes (CI1, CI2, CI3, CI4, CI5, CI6, and BM6) in the containment event trees result in failure of the containment building by a variety of failure modes. Each of these failure modes results in a particular radiological release category. For those paths that do not have a negative response at any of the seven nodes, the path will eventually result in no failure of the containment. The containment event trees, therefore, link damage states to a range of possible containment failure modes via the various paths through the tree. For a given tree, each path ends in a conditional probability (CP) of occurrence and these CPs should sum to unity. The quantification of an event tree is the process by which all the

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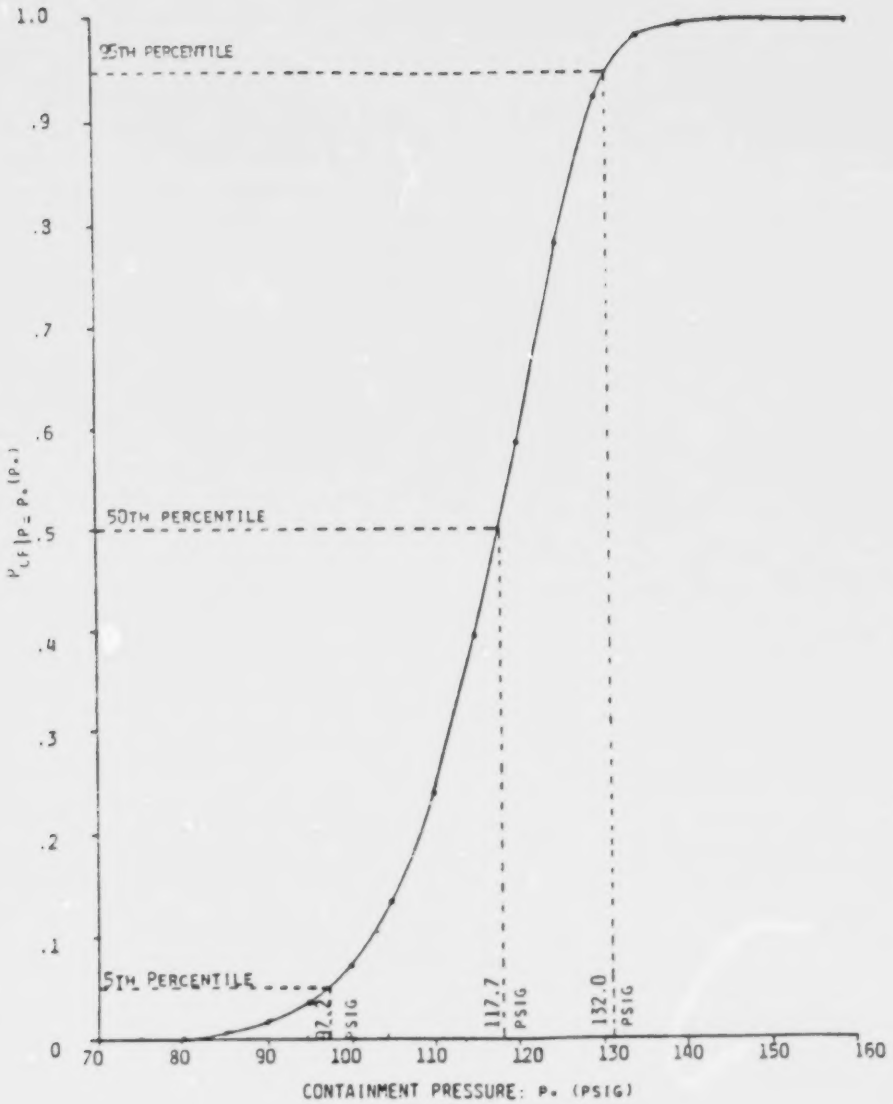

Figure 3.2 Containment Failure Pressure Distribution for Millstone-3

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Table 3.7 Summary of Containment Event Tree Time Frames and Nodal Questions[5]

Time Frame I:	Accident Initiation $\leq$ t < Core Degradation	
	CI1	Is the containment intact?
Time Frame II:	Core Degradation $\leq$ t < Significant Debris Accumulation in Lower Plenum	
	NB2	Does the hydrogen not burn?
	CI2	Does the containment remain intact?
Time Frame III:	Significant Debris Accumulation $\leq$ t < Vessel Failure in Lower Plenum	
	CD	Is the core melt incoherent?
	NB3	Does the hydrogen not burn?
	CI3	Does the containment remain intact?
Time Frame IV:	Vessel Failure $\leq$ t $\leq$ Complete Depressurization	
	QUE	Is the core debris quenched?
	NB4	Does the hydrogen not burn?
	CI4	Does the containment remain intact?
Time Frame V:	Complete Depressurization < t $\leq$ 4 Hr* After Vessel Failure	
	CD5	Is the debris coolable?
	NB5	Does the hydrogen not burn?
	CI5	Does the containment remain intact?
Time Frame VI:	4 Hr After Vessel Failure < t < One Day	
	CD6	Is the debris coolable?
	NB6	Does the hydrogen not burn?
	CI6	Does the containment remain intact?
	BM6	Does the basemat remain intact?

*The estimated time to boil off the accumulator water from the lower reactor cavity is 4 hours.[5]

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paths are combined to give the conditional probabilities of the various release categories. In the MPSS-3, thirteen release categories were used for the quantification process as summarized in Table 3.8. Note that one of these release categories (namely, M12) corresponds to no containment failure. Fission product release for this category would, therefore, be via normal design (allowable) leakage paths in the containment building.

The quantification of the Millstone-3 containment event trees was a significant task, and it was necessary to use a computer code, ARBRE, to group the various path probabilities into the thirteen release categories.[5] The containment matrix 'C' is a concise summary of the quantification process.

Table 3.9 is a reproduction of the C-matrix for the MPSS-3.[5] It lists the conditional probabilities of the release categories given the containment response class. A simplification to the C-matrix is obtained in Table 3.10 by disregarding all of the very low probability values.

3.5 BNL/NRC Reassessment of the C-Matrix

Staff of Brookhaven National Laboratory (BNL) together with the NRC staff, have reviewed those aspects of the MPSS-3 related to the containment response and failure modes. However, they have not performed independent confirmatory calculations of accident progression and containment response. Instead, adjustments to the MPSS-3 have been made based on the experience gained from the previous in-depth reviews of similar risk studies (i.e. Zion[1] and Indian Point[3] Probabilistic Safety Studies).

The mode and timing of containment failure cannot be calculated with great accuracy. Judgments must be made about the nature of the dominant phenomena and about the magnitude of several important parameters. Furthermore, the codes and methods used for these calculations are approximate and do not model all of the detailed phenomena. Fortunately, risk estimates are not sensitive to minor variations in failure mode and timing. It is important, however, to properly characterize the major attributes of failure mechanisms: (1) whether the failure is early or late, (2) whether it is by overpressurization, bypass or basemat melt-through and (3) whether or not radionuclide removal systems are effective.

3.5.1 Conditional Probability of Release Categories

Our reassessment of the containment response and failure mechanism was based on our general understanding of the accident phenomenology and the containment design. The resulting simplified C-matrix shown in Table 3.11 includes only those failure modes which would contribute significantly to risk. The phenomena of interest may be summarized as follows:

In-Vessel Steam Explosions (M2B), which result in direct containment failure are believed to be highly unlikely. Although explosions in the reactor vessel lower plenum are highly probable, the resulting mechanical energy would be limited by the fraction of the core which could participate in a single explosion and by the efficiency of the process. In recent PRA reviews,

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Table 3.8 Notation and Definitions for Release Categories

Release Category	Description
M1A	Containment Bypass, V-sequence
M1B	Containment Bypass, SGTR
M2	Early Failure/Early Melt, No Sprays
M3	Early Failure/Late Melt, No Sprays
M4	Containment Isolation Failure
M5	Intermediate Failure/Late Melt, No Sprays
M6	Intermediate Failure/Early Melt, No Sprays
M7	Late Failure, No Sprays
M8	Intermediate Failure With Sprays
M9	Late Failure With Sprays
M10	Basemat Failure, No Sprays
M11	Basemat Failure With Sprays
M12	No Containment Failure

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Table 3.9 Containment Failure Probabilities (C-Matrix) from the MPSS-3[5]

Containment Class	M1A	M1B	M2	M3	M4	M5	M6	M7	M8	M9	M10	M11	M12
1	-	-	2.87E-4	-	2E-4	-	6.16E-1	2.9E-1	-	-	9.34E-2	-	1.32E-5
2	-	-	1.15E-3	-	2E-4	-	6.17E-2	8.9E-1	-	-	4.6E-2	-	1.40E-5
3	-	-	3.07E-4	-	2E-4	-	5.38E-1	3.5E-1	-	-	1.13E-1	-	1.59E-5
4	-	2E-4	-	1.0E-4	2E-4	4.36E-3	-	9.0E-1	-	-	9.83E-2	-	1.98E-5
5	-	-	-	2.3E-4	2E-4	9.48E-3	-	8.0E-1	-	-	1.96E-1	-	2.94E-5
6	-	2E-4	2.9E-4	-	2E-4	-	-	-	6.37E-4	1.18E-4	-	5E-2	9.48E-1
7	-	-	-	2.9E-4	2E-4	6.38E-4	-	9.98E-1	-	-	-	9.89E-4	3.51E-5
8	-	-	2.4E-4	-	2E-4	-	-	9.98E-1	6.77E-4	-	-	9.77E-4	2.16E-5
9	-	2E-4	2.9E-4	-	2E-4	-	-	-	6.63E-4	1.19E-4	-	9.89E-1	1E-2
10	-	-	2.1E-4	-	2E-4	1.1E-3	-	9.89E-1	-	-	9.88E-3	-	1.1E-5
V	1.0	-	-	-	-	-	-	-	-	-	-	-	-
V_2	-	1.0	-	-	-	-	-	-	-	-	-	-	-
V_3	-	-	-	-	1.0	-	-	-	-	-	-	-	-

Table 3.10 Simplified C-Matrix Based on the MPSS-3

Containment Class	Plant Damage States	M1A	M1B	M4	M5	M6	M7	M9/M10/M11	M12
1	AE					0.62	0.29	0.09	
2	SE					0.06	0.89	0.05	
3	AL					0.54	0.35	0.11	
4	TE						0.90	0.10	
5	SL					0.01	0.79	0.20	
6	_ _C							0.05	0.95
7,8	_ _C'						1.0		
9	_ _C"								
10	S'E,S'L							0.99	0.01
V		1.0					0.99	0.01	
V_2			1.0						
V_3				1.0					

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we have assigned a conditional probability of 10^{-4} to direct containment failure due to this mechanism.[1-3] At this level, steam explosions would have a negligible effect on risk, and consequently tney are not included in the simplified C-Matrix (Table 3.11).

Failure to Isolate Containment (M4) is considered a relatively unlikely event for a subatmospheric containment, because the plant is required by technical specification to maintain a partial vacuum during all modes of operation except cold shutdown and refueling. Thus, there is a high probability that all sizable penetrations to the containment will be isolated prior to a severe accident. While we are not convinced the probability is as low as the applicant assumes ($2x10^{-4}$), we are confident that it is low enough to be a minimal contributor to risk. Therefore, we do not include this failure mode (M4) in the simplified containment matrix (Table 3.11). (The M4 failure due to the V_3 containment response class will be explained in Section 3.6 below.)

Early Containment Failure (M2A and M3) following core melt can result from rapid steam production as the core enters the reactor cavity (steam spike), or from a prompt hydrogen burn. Simple adiabatic calculations show that the steam spike from complete and rapid quench of the core would produce a peak pressure well below the containment failure point (refer to Fig. 3.2). Similarly, assuming 1000 pounds of in-vessel hydrogen production, a complete adiabatic hydrogen burn at the time of vessel failure would also fall well below the failure pressure. A combination of a steam spike and hydrogen burn would be very unlikely because the steam tends to inert the containment atmosphere at high steam mole fractions (>0.50) and prevent H_2 burning. Our calculations indicate that even this improbable occurrence would not be likely to fail containment.

Early failure could also conceivably result from direct heating due to a rapid dispersal of the core debris throughout containment in the form of aerosols. The dispersal could only be caused by the high primary system pressures that would exist at vessel failure for a number of transient sequences. The aerosols could rapidly pressurize containment by direct heat exchange and concomitant chemical reactions. Scoping calculations performed by the Containment Loads Working Group (CLWG) showed that a very severe challenge to the containment integrity could result provided 25 percent of the core mass were converted to aerosol.[12] However, no consensus could be reached among the CLWG analysts as to the credibility of this parameter value, and this failure mode is still under investigation by the CLWG. For the purpose of this review, we do not assume any direct heating failure will occur. Millstone-3 has a somewhat larger volume, and higher failure pressure than considered by the CLWG. Furthermore, the design of the Millstone cavity would tend to suppress the dispersal of core debris beyond the cavity boundaries. Our assessment is that this mode of failure is not likely at Millstone-3. This issue is discussed further as part of the sensitivity analysis in Section 6.1.

For the reasons outlined above, we have concluded that early overpressure failure (M2A and M3) has a very low likelihood.

Intermediate Overpressure Failure (M6 and M6S) can occur due to combustion of hydrogen. Some time between four and sixteen hours after vessel

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Table 3.11 BNL/NRC Containment Matrix†

Containment Response Class	M1A	M1B	M4	M6	(M6S)	M7	M10/M11	M12
1				0.62		0.29	.09	
2					0.34	0.03(1.0)*		0.63
3				0.54		0.35	0.11	
4						0.90	0.1	
5						0.80	0.2	
6							.05	0.95
7,8						1.0		
9							0.99	0.01
10						0.99	0.01	
V	1.0							
V_2		1.0						
V_3			1.0					

†Matrix elements with low probabilities and with low contributions to overall risk have been deleted from the C-Matrix. This includes steam explosions (M2B) and early overpressure failures (M2A and M3).

*The conditional failure probability of the M7 release category following a seismically induced station blackout (SE) is 1.0. For non-seismic station blackouts, the conditional probabilities for M6S, M7, and M12 are 0.34, 0.03, and 0.63, respectively.

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failure, the accumulated hydrogen production from all sources is expected to reach approximately 2,860 pounds, the maximum amount that could be burned by the oxygen in containment. The wide spread in estimated timing is due to uncertainties in the rate at which core-concrete interaction progresses. Detailed thermal-hydraulic calculations by the applicant and calculations performed as part of our review have shown that a hydrogen burn of this magnitude could fail containment only for a narrow band of conditions in which the steam concentration is high enough to produce a sufficient containment background pressure, but not sufficient to inert the containment. There are only two scenarios in which this is believed to be attainable: (1) the AE sequence and (2) station blackout.

In the AE sequence, the applicant's thermal hydraulic calculations show that the containment atmosphere becomes deinerted less than an hour after core melt and reaches peak flammability about three hours later with a background pressure of 50 psia. The applicant's estimate of a high (0.62) probability for intermediate failure (M6) is reasonable in view of these conditions. The fact that it is less than unity would reflect the uncertainty in the burn pressure spike and also the possibility that the burn would occur before the hydrogen concentration reached the containment failure threshold.

A similar analysis for the SE sequence showed the atmosphere to be marginally inerted throughout the accident. As a measure of conservatism, and to account for uncertainties, the applicant has assumed a 0.06 probability for hydrogen burn failure at intermediate time (M6) in SE sequence.

In the NRC/BNL review, the SE sequence is dominated by station blackout sequences resulting in pump seal LOCA, in which containment sprays are not recovered until six or more hours after core melt. We postulate that a large hydrogen burn is likely to occur due to deinerting of containment in the period beyond six hours. If deinerting results from late resumption of spray operation, we estimate a 50 percent probability of containment failure. However, spray operation is estimated to greatly reduce the suspended aerosol concentration, thereby leading to reduced consequences. The corresponding release category is labelled M6S.

Recent calculations by Sandia National Laboratory[9] indicate that deinerting due to natural condensation processes is also likely. The containment failure probability in the event of condensation deinerting is estimated by the NRC and BNL to be in the vicinity of 10 percent, because the energy efficiency of the burn would be suppressed by the presence of high steam concentrations. Condensation deinerting would not yield the large source term reductions experienced with spray deinerting. Consequently, this failure mode was assigned to the M7 release category.

The release category probabilities quoted in Table 3.11 for the SE plant damage state were derived based on the relative probabilities of spray deinerting and condensation deinerting, along with the conditional containment failure probability for each. Implicit in the numbers is the assumption that some form of AC power is recovered within a day after vessel failure. This assumption is reexamined in Section 6 below. The parenthetical value of 1.0 for the M7 release probability was used for SE events caused by earthquakes.

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It reflects our assumption that containment overpressure failure is inevitable for seismically induced station blackouts.

Late Overpressure Failure (M7) can occur due to steam production in a wet cavity or noncondensible gas production in a dry cavity. For sequences in which early and intermediate failure is not expected to occur, and in which the containment spray recirculation is inoperable, we conclude that the containment will fail due to late overpressurization. These sequences include AE, TE, SE sequences other than station blackout, sequences in which only quench spray operates (AEC', SEC'...), and instrument tube LOCA's (S'L).

It is possible that late overpressure failure will be precluded by slow depressurization of containment due to leakage. Such enhanced leakage could result from containment penetration seal degradation due to the high containment temperatures. A methodology for assessing this type of leakage has been developed by the NRC Containment Performance Working Group (NUREG-1037) and applied to several plants. Neither the NRC nor the applicant has made such an analysis of leak-before-failure specifically for the Millstone-3. Consequently, for the purpose of this review, the reduction in late overpressure failure probability due to this mechanism has not been accounted for.

The Overpressure Failure with Sprays (M8 and M9), Basemat Melt-through (M10 and M11) or No failure (M12) would result from sequences in which spray operation or recirculating spray operation would occur, leading to reduced fission product release and negligible off-site consequences.

Three Containment Bypass modes were identified: the interfacing system LOCA (V), the steam generator tube rupture (V_2) and the seismically induced crane wall failure (V_3). Unique release categories were defined for the interfacing systems LOCA (M1A) and steam generator tube rupture (M1B). The crane wall failure is postulated to cause a large LOCA with failure of ECC and a simultaneous breach of the containment. The V_3 sequence is assigned to the M4 release category, in which a large fraction of all fission products are postulated to be released to the environment over a two-hour period immediately following core melt. This is a reasonable choice because the M4 release category represents a large early release with relatively low release energy.

3.5.2 Release Category Frequencies

Based on the containment class frequencies in Table 3.5 and the containment failure matrix of Table 3.11 the frequencies for each release category were calculated and are listed in Table 3.12.

3.6 Accident Source Terms

In this section the approach used in the MPSS-3 to determine the fraction of fission products originally in the core which can leak to the outside environment will be outlined. The fission products released to the environment as calculated by this approach for each release category will also be discussed.

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Table 3.12 Frequencies of Risk-Significant Release Categories (Per Reactor Year)

Release Category	Frequency (Per Reactor Year)			
	Based on Dames & Moore Seismic		Based on SHCP Seismic	
	Normal Evacuation (Internal, Fire and Seismic <.5g)	Impaired Evacuation (Seismic >.5g)	Normal Evacuation (Internal, Fire and Seismic <.5g)	Impaired Evacuation (Seismic >.5g)
M1A	8E-7	-	8E-7	-
M1B	2.5E-6	-	2.5E-6	-
M4	1.6E-8	8.4E-8	3.0E-7	3.7E-6
M6	4.8E-8	2.6E-7	6.8E-7	5.5E-6
M7	8.6E-6	3.1E-6	4.6E-5	6.3E-5
M6S	4.6E-6	-	4.6E-5	-

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As in the Reactor Safety Study (RSS)[10] methodology, the CORRAL-II code is the most important tool for determining the fission product leakage to the environment. Input to this code is obtained from the thermal-hydraulic analysis carried out for the containment atmosphere. In addition, the time-dependent emission of fission products is provided as input to the code. The fission product release is divided up into the customarily used phases, i.e., Gap, Melt, and Vaporization releases. The time dependence of these phases is determined by the timing of core heatup, primary system failure, and core/concrete interaction. In all, thirteen release categories were estimated ranging from the containment bypass sequence (V-sequence) to the no-fail sequence (Table 3.13).

The M1A release fractions and timing are identical to the PWR2 release in the RSS. The release M1B, which corresponds to a steam generator tube rupture which progresses to a core melt, was determined by dividing PWR2 or M1A by ten. Noble gases and organic iodine are not subject to this reduction in release.

The source terms for overpressure failure were all based on plant specific calculations with CORRAL-II. For early failure without sprays (M2 and M3) a large fraction of the fission products that are not still trapped in the core debris are released to the environment. Not surprisingly, the calculated releases for the failure to isolate containment (M4) are similar in magnitude. For intermediate (M5 and M6) and late overpressurization (M7) without sprays, the iodine concentration is greatly reduced, but all other species remain fairly constant or increase due to additional releases from the core-concrete interactions. With the exception of iodine, very little credit is taken for source term reduction due to residence time in the containment.

On the other hand, the intermediate and late overpressure containment failures with sprays (M8 and M9) have significantly lower release fractions compared with the analogous cases without sprays. With the exception of noble gases, all fission products were estimated to be significantly reduced by the operation of sprays.

Fission product releases for the basemat melt-through with and without sprays (M11 and M10) were taken directly from the values used in the RSS. They represent a decontamination factor of 1000 due to soil filtration.

The M12 sequence represents the case in which the sprays are operational and the containment remains intact. A leak rate of 0.9 volume percent per day is assumed. Releases are extremely low because of the assumed low leak rate and the reduced levels of suspended radioactivity in the containment due to spray operation. No credit was taken for fission product removal by the Supplementary Leak Collection and Release System.

The release fractions in Table 3.13 do not reflect all mechanisms of source-term attenuation. Retention of fission products in the primary system was not credited. Furthermore, the enhancement of gravitational settling in containment due to aerosol agglomeration was not included. To account for these factors and their associated uncertainties, the MPSS-3 employed the method of discrete probability distributions (DPD) for those release categories in which CORRAL predicts large releases of fission product aerosols.

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Table 3.13 MPSS-3 Release Category Summary[5]

Release Category	Release Start Time (hrs)	Release Warning Time (hrs)	Release Duration (hrs)	Release Energy (Btu/hr)	Fission Product Release Fraction							
					Xe-Kr	OI	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
M-1A	2.5	1.0	1.0	20 E6*	9E-1	7E-3	7E-1	5E-1	3E-1	6E-2	2E-2	4E-3
M-1B	2.5	1.0	1.0	20 E6	9E-1	7E-3	7E-2	5E-2	3E-2	6E-3	2E-3	4E-4
M-2	0.75	0.2	2.0	150 E6	7E-1	5E-3	5E-1	6E-1	2E-1	7E-2	2E-2	3E-3
M-3	6.0	0.5	2.0	190 E6	8E-1	5E-3	5E-1	6E-1	2E-1	8E-2	3E-2	3E-3
M-4	0.2	0.0	2.0	70 E6	9E-1	6E-3	2E-1	6E-1	5E-1	7E-2	5E-2	7E-3
M-5	8.3	4.1	0.5	450 E6	9E-1	6E-3	1E-2	5E-1	5E-1	5E-2	4E-2	6E-3
M-6	4.3	4.1	0.5	440 E6	9E-1	6E-3	1E-2	5E-1	5E-1	5E-2	4E-2	7E-3
M-7	20.1	16.0	0.5	540 E6	9E-1	6E-3	9E-3	3E-1	3E-1	3E-2	2E-2	4E-3
M-8	4.5	4.0	0.5	22 E6	9E-1	7E-3	8E-3	1E-5	1E-5	1E-6	1E-6	2E-7
M-9	21.0	20.0	0.5	22 E6	9E-1	6E-3	2E-3	2E-6	1E-6	2E-7	9E-8	1E-8
M-10	95.0	80.0	10.0	NA	3E-1	2E-3	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
M-11	95.0	80.0	10.0	NA	6E-3	2E-5	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7
M-12	0.5	0.0	5.0	NA	1E-3	9E-6	6E-6	1E-6	9E-7	2E-7	8E-8	1E-8

*20E6 = 20 X 10^6 = 20,000,000

Table 3.14 MPSS-3 Release Category DPDs[5]

Release Category	Probability (P) Associated With Release Fraction (F)				
	F/1	1/2*	1/4	1/10	1/100
M1A	0.17	0.55*	0.28	0	0
M2	0.25	0	0.25	0.50	0
M3	0.0	0	0.06	0.63	0.31
M4	0.40	0.60	0	0	0
M5	0.0	0.0	0.05	0.64	0.31
M6	0.11	0.14	0.27	0.48	0
M7	0	0	0	0.11	0.89

*For release category M1A, the probability (P) that the actual fission product releases will be one half (F=1/2) of the values calculated by CORRAL is estimated to be 0.55.

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In this method, the actual release fractions for a given release category can assume values which are a fraction (F) of the values given in Table 3.13. The allowed fractions are 1, 1/2, 1/4, 1/10, and 1/100. A probability (P) is associated with each F, and the probabilities are different for each release category (Table 3.14). For example, in a failure to isolate containment (M4), there is an assumed 40% probability that F is equal to unity, and a 60% probability that F is 1/2. This small reduction in fission product release reflects an assumed retention of fission products in the primary system, but very little effect of agglomeration. For late failure without sprays (M7), agglomeration is assumed to play a significant role, and the source term would be reduced by a factor of 10 to 100. The values of F and P are based largely on engineering judgment.

3.7 BNL/NRC Reassessment of Radiological Source-Terms

There are three significant areas in which we questioned the release fractions used in the MPSS-3. These are the release fractions for the bypass sequences, the iodine release for the overpressurization failure sequence (M5, M6 and M7), and release timings for certain sequences.

It is noted that the WASH-1400 PWR2 estimated release used in the MPSS-3 report for the release category M1A was defined for sequences other than the interfacing system LOCA (Event V). The Oconee-RSS-MAP[11] study produced a set of estimated releases specifically defined for the characteristics of a V sequence. Because the RSS-MAP source term estimate includes virtually no credit for fission product retention in the auxiliary building, differences between Oconee and Millstone-3 design are not important. Consequently, we decided to substitute the Oconee release fractions for the MPSS-3 values in release category M1A (see Table 3.15).

MPSS-3 does not give any basis for setting the M1B release fractions equal to one tenth of the PWR2 releases with the exception of noble gases and organic iodine. It is reasonable to assume that some fraction of the fission products would be deposited in the primary system piping, steam generator internals and steam separators/dryers. Furthermore it appears that no CORRAL-II calculations have been performed for the tube rupture accident. Fortunately, for the large majority of core melt sequences from tube ruptures, RRAB estimates that there is no leakage through the relief valves. Further, if the release is through the condenser, we can assume considerable deposition of fission products. Consequently, the applicant's values are used in the present calculations.

The licensee calculated two sets of source terms, one based on the assumption of elemental iodine, and the other based on the more commonly accepted model of iodine chemically bonded with cesium. Although the MPSS-3 risk calculations were based on the cesium iodide model, we chose to employ the elemental Iodine model in order to be more consistent with WASH-1400 methods. However, comparisons of Table 3.13 with other studies performed with WASH-1400 methodology[10] show discrepancies in the iodine releases for the M5, M6 and M7 release categories. In References [3] and [12], the iodine releases for late overpressure failure estimated for the Indian Point reactors were an order of magnitude higher than the MPSS-3 results (Table 3.16). The

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Table 3.15 BNL/NRC Release Category Summary (Internal and External Events)

Category	Release Duration (hrs)	Release Energy (Btu/hr)	Fission Product Release Fractions							
			Xe-Kr	OI	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
M-1A	1.0	0.5 E6	1E-0	7E-3	4.8E-1	7.9E-1	4.4E-1	9E-2	4E-2	6E-3
M-1B	1.0	0.5 E6	9E-1	7E-3	7E-2	5E-2	3E-2	6E-3	2E-3	4E-4
M-2A	2.0	150 E6	7E-1	5E-3	5E-1	6E-1	2E-1	7E-2	2E-2	3E-3
M-2B	0.5	520 E6	9E-1	-	7E-1	4E-1	4E-1	5E-2	4E-1	3E-3
M-3	2.0	190 E6	8E-1	5E-3	5E-1	6E-1	2E-1	8E-2	3E-2	3E-3
M-4	2.0	70 E6	9E-1	6E-3	2E-1	6E-1	5E-1	7E-2	5E-2	7E-3
M-5	0.5	150 E6	9E-1	6E-3	1E-2	5E-1	5E-1	5E-2	4E-2	6E-3
M-6	0.5	150 E6	9E-1	6E-3	1E-2	5E-1	5E-1	5E-2	4E-2	7E-3
M-6S	0.5	70 E6	9E-1	-	7E-3	1E-3	1E-3	1E-4	9E-5	1E-5
M-7	0.5	150 E6	9E-1	6E-3	9E-3	3E-1	3E-1	3E-2	2E-2	4E-3
M-8	0.5	22 E6	9E-1	7E-3	8E-3	1E-5	1E-5	1E-6	1E-6	2E-7
M-9	0.5	22 E6	9E-1	6E-3	2E-3	2E-6	1E-6	2E-7	9E-8	1E-8
M-10	10.0	n/a	3E-1	2E-3	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
M-11	10.0	n/a	6E-3	2E-5	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7
M-12	5.0	n/a	1E-3	9E-6	6E-6	1E-6	9E-7	2E-7	8E-8	1E-8

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Table 3.16 Comparison of Estimated Release Fractions for Intermediate and Late Overpressurization (No Sprays); Millstone-3 Versus Indian Point

Fission Product Group	Release Fraction			
	Sequence			
	MPSS-3* M5	MPSS-3* M7	IPS [12]† TMLB'-δ	IPPSS [3]† 2RW
Xe-Kr	9E-1	9E-1	9.6E-1	1E0
IO+I	1.6E-2	1.5E-2	1.05E-1	9.3E-2
Cs-Rb	5E-1	3E-1	3.4E-1	2.6E-1
Te-Sb	5E-1	3E-1	3.8E-1	4.4E-1
Ba-Sr	5E-2	3E-2	3.7E-2	2.5E-2
Ru	4E-2	2E-2	2.9E-2	2.9E-2
La	6E-3	4E-3	4.9E-3	1.0E-2

*These figures represent the MPSS-3 elemental iodine model.[5]

†Fission product release fractions for late overpressure failure as estimated in the Indian Point Probabilistic Safety Study (IPPSS) and the NRC staff review of Indian Point (IPS).

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releases of all other radionuclides were of comparable magnitude. This is due to the fact that we are using source terms characteristic of gaseous iodine. We have not adjusted the MPSS-3 release fractions. However, we have performed a consequence calculation which shows that an order of magnitude increase in Iodine releases would have a minimal impact on risk.

The energy of release for release categories M5 through M7 is high when compared to similar sequences (overpressurization failure) in other PRAs. The MPSS-3 release energy is more characteristic of a steam explosion release (α) as evaluated in the RSS. This value is a function of the failure pressure and the assumed rupture size. Furthermore, it has been found that a high energy of release lifts the plume to a higher altitude than a lower release energy. The overall effect of this is to disperse the plume, and thus reduce the concentration of the dose received by the surrounding population. This reduction in the dose affects consequences which are a function of a threshold dose, i.e., early fatalities. In view of the uncertainty and the possible effect on early fatalities, we have reduced the release energy from 450×10^6 to 150×10^6 BTU/hr as shown in Table 3.15 in our consequence analyses. The reduced release energy is consistent with values quoted in the Reactor Safety Study for overpressure failure.

Due to the significance of the release time of the early release categories for generating early fatalities, and also the importance of the warning time on the effectiveness of evacuation, the release and warning times quoted in the MPSS-3 were reevaluated for each response class. The results of the reevaluations, based on the dominant sequence in each response class, are summarized as follows:

For the M1A and M1B release categories, the MPSS-3 release time and warning times are acceptable.

Early and intermediate overpressure failures result primarily from hydrogen burns. The release times for categories M2, M3, M5, M6 and M8 are calculated as the time when containment atmosphere becomes flammable; otherwise, they are set to the vessel failure time for a given containment response class. Estimates of the time of flammability were taken from Section 4.4 of the MPSS-3.[5] The warning time is defined as the time after core melt starts to the time of release. For M2B and M4 the release is assumed to occur following vessel failure, and the warning time is the time following the core melt to the time of radiological release in our consequence analyses.

For M7 and M9 through M12 the MPSS-3 release and warning times are acceptable since their influence on acute fatalities is negligible.

In the consequence analysis, only one value of the warning time was used for each failure mode (Table 3.17). These values were chosen to represent the dominant sequence for each failure mode.

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Table 3.17 Release Times, Release Durations and Warning Times Assumed for Risk Significant Release Categories in the BNL/NRC Consequence Analysis

Failure Mode	Release Time (hr)	Release Duration (hr)	Warning Time (hr)
M1A	2.5	1.0	1.0
M1B	2.5	1.0	1.0
M4	1.5(I)*	2.0	0.25(I)
	2.5(E)*		1.0(E)
M5	5.6	0.5	0.4
M6	0.8	0.5	0.2
M6S	11.5	0.5	5.9
M7	20.1	0.5	16.0

*Times given are for internal (I) and external (E) events.

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The DPD methodology (Section 3.6, Table 3.14) has been examined and it was concluded that it should not be factored into the release fractions used for the present evaluation. Fission product retention in the primary system and aerosol agglomeration in containment are credible mechanisms for fission product attenuation, and are currently under study by the NRC Accident Source Term Program Office (ASTPO). Because the ASTPO evaluation of the existence and magnitude of these mechanisms was not complete at the time we performed this assessment, there was no sound basis for assuming a reduction in radiological releases for all release categories. Even if the ASTPO evaluation was available when this assessment was performed it is unlikely the DPD methodology would have been accepted. This methodology is subjective in nature. We believe that reductions in source term estimates should be based on explicit mechanistic calculations. It is recognized that the decision not to factor in the DPD's represents a conservative approach to the source-term.

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4. CONSEQUENCE CALCULATION METHODOLOGY AND THE MILLSTONE-3 SITE MODEL

4.1 Introduction

The NRC staff has estimated the consequences and risks of potential severe accidents at Millstone-3. The first part of this work was done for the Millstone-3 Final Environmental Statement (NUREG-1064). A more complete discussion of the methodology is found in that report, as well as a discussion of general characteristics of accidents, fission product characteristics, exposure pathways, health effects, post-accident exposure avoidance, accident experience, an evaluation of consequences assuming evacuation and no evacuation (just early relocation) and a critique of the applicant's consequence analysis (Section 5.9.4 and Appendices F, L, M and N of NUREG-1064).

The exposure pathways considered here are restricted to those that start with a release to the atmosphere because the NRC staff concluded that the risk contribution of the liquid pathway at Millstone-3 is small in comparison to the risk posed by airborne pathways (p. 5-50 of NUREG-1064).

4.2 Probabilistic Assessment of Severe Accident Consequences

The calculative methodology used to estimate the conditional consequences of various severe releases is essentially as described in the Reactor Safety Study,[10] but includes improvements in the assessment methodology that were developed after publication of the RSS. Potential accidents initiated by fires and external events have been included in this analysis, but the effects of possible sabotage have not been included.

In general, there is a very large number of potential accident sequences. These are reduced or combined to represent the few most important to risk. Depending on containment behavior, these accident sequences are predicted to lead to several different releases as defined previously. Simplified descriptions of the releases of radioactive material in a form useable by the consequence calculation code, are called release categories (see Chapter 3).

Tables 3.15 and 3.17 provide information used in the NRC staff's consequence assessment for each specific release category. The information includes time estimates from termination of the fission process during the accident until the beginning of release to the environment (release time), duration of the atmospheric release, warning time for offsite evacuation, and estimates of the energy associated with the release, height of the release location above the ground level, and fractions of the core inventory of eight groups of radionuclides in the release.

The magnitudes (curies) of fission products released to the atmosphere for each accident sequence or release category are obtained by multiplying the release fractions shown in Table 3.15 by the amounts that would be present in the core at the time of the hypothetical accident and by depletion factors that result from in-plant radioactive decay before the release to the environment. The core inventory of radionuclides used is based on a core thermal

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power level of 3579 MWt, the same power level used in the Millstone-3 Final Safety Analysis Report for accident analysis (see Table 5.13 of NUREG-1064). From the eight groups listed in Table 3.15, 54 nuclides were used to represent those that are potentially major contributors to the health and economic effects of severe accidents.

Many estimated radiological consequences of the postulated release categories were calculated using the computer code CRAC, which is based on the RSS consequence model (see NUREG-0340 and NUREG/CR-2300). Three types of consequences (early fatalities, latent fatalities and person-rem) were used as surrogates for all severe accident consequences (including numerous impacts not directly treated by CRAC). CRAC has been adopted and modified to be site-specific and to better model evacuation. The site-specific data input to CRAC comprises the "site model," as it is often called. The nature of these data and the way they were used is summarized below.

4.3 Use of the Millstone-3 Site Model

Information describing the Millstone-3 site was independently determined or checked by the NRC staff.

These data include:

(1) meteorological data for the site representing a full year (1981) of consecutive hourly measurements and seasonal variations

(2) projected population for the year 2010 extending throughout regions of 80-km (50-mile) and 563-km (350-mile) radius from the site

(3) the habitable land fraction within a 563-km (350-mile) radius

(4) land-use statistics on a statewide basis, including farm land values, farm product values including dairy production, and growing season information, for the State of Connecticut and each surrounding state within the 563-km (350-mile) region

For the region beyond 563 km (350 miles), the U.S. average population density was assumed.

The calculation was extended out to 3200 km (2000 miles) from the site to account for the residual radionuclides that would remain in the atmosphere at large distances with rain assumed in the interval between 563 km and 3200 km to deplete the plume of all non-noble-gas inventory. To sample the dispersion conditions as they vary during a year, calculations were performed assuming the occurrence of each release category at each of 91 different "start" times distributed throughout a 1-year period. Each calculation used site-specific hourly meteorological data and seasonal information for the period following each start time.

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Two different sets of assumptions about offsite emergency response were used, depending on whether or not the postulated accident was initiated by a severe earthquake (the results of a third emergency response scenario were presented in the DES as a sensitivity study). The basic premise behind these assumptions is that under normal or even fairly bad weather conditions, early evacuation and relocation of people would considerably reduce both cloud and ground exposure; while a severe earthquake would prevent early evacuation for most people. The early evacuation/relocation (abbreviated as Evac-Reloc) assumptions were used for those accidents initiated by internal events, fire, and earthquakes of low to moderate severity (instrumental peak ground acceleration less than 0.5g), equivalent to Modified Mercalli Intensity Scale VIII. The second set was used solely for accidents initiated by severe earthquakes. The sets of assumptions for both scenarios are shown in Table 4.1.

The value of delay time in Table 4.1 is consistent with the NRC requirement regarding prompt notification of the public of the emergency, and the time people would take preparing for evacuation after being notified of the emergency during normal to moderately adverse conditions such as snow, ice, hurricane, and low to moderately severe earthquakes. The values of delay time before evacuation and effective evacuation speed used in the NRC staff analysis are assumed only to be average values. For areas beyond 16 km (10 miles) the parameters selected reflect the assumption that an extension of emergency response would occur during a large accident and people would be advised to leave areas that would be considered to be highly contaminated (see below for criterion); that is, people would relocate. Relocation of the public from the highly contaminated areas beyond 16 km (10 miles) is assumed to take place 12 hours after plume passage. The criterion for this relocation is whether the projected 7-day ground dose to the total bone marrow, as projected by field measurements, would exceed 200 rems (which is only slightly above the average threshold exposure for potential early fatality with minimal medical treatment); otherwise, people in highly contaminated areas are assumed to be relocated within 7 days.

The second set of parameters reflects a radiological emergency response situation hampered by a severe type of external event, such as a severe regional earthquake, which would seriously limit the ability to evacuate and would also eliminate or reduce the shielding protection that the public would otherwise experience. However, relocation of the public from highly contaminated areas 24 hours after plume passage is assumed. The criterion for this relocation is the same as in the first set of assumptions, but relocation is assumed to extend outward from the site exclusion area boundary (503 m, as opposed to the 16-km (10-mile) EPZ boundary); otherwise, people are assumed to be relocated within 7 days. The offsite emergency response mode characterized by this second set of assumptions is designated Late Reloc.

In addition to the two types of short-term emergency response described above, a long-term response (common to both scenarios above) is modeled. The long-term environmental protective actions to reduce chronic exposure include complete interdiction of food; decontamination of food, property, or land; and/or temporary interdiction of food, property, or land followed by decontamination.

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Table 4.1 Emergency Response Assumptions for Millstone-3

Emergency Response Scenario	Evacuation				Relocation			Shielding Protection Factors†	
	Area Evacuated (miles from the plant)	Delay Time After Warning (hr)	Effective Evacuation Speed (mph)	Effective* Evacuation Distance (mi)	Distance Beyond Which Relocation Occurs (miles from the plant)	Time of Relocation (hr) after plume passage	Relocation** Dose Criterion (7-day projected bone marrow dose) (rem)	During Evacuation (plume dose/ ground dose)	Other Times (plume dose/ ground dose)
Evac-Reloc	10	1	2	15	10	12	200	1/0.5	0.75/0.33
Late-Reloc	N/A	N/A	N/A	N/A	0	24	200	N/A	1.0/0.5

N/A = Not Applicable.

*An artificial parameter used only to represent an effective path length for each evacuee over which radiation exposure to the evacuee is calculated in the CRAC code.

**People would be relocated from any area where the radiation level is high enough to give a 200 rem dose to the bone marrow over a seven-day period.

†Defined as ratio of dose with protection to dose without protection. During evacuation, automobiles are expected to give no protection from plume dose, but some protection (0.5) from ground dose. For other times, the protection factors represent normal activities for people and were extracted from WASH-1400, Appendix VI, Tables VI 11-13. During severe earthquakes buildings may not remain habitable and would thus provide no protection from plume dose. However, the buildup of mud and debris would provide some protection from ground dose.

The consequences of severe accidents in the Millstone-3 reactor initiated by plant internal causes, fires, and low to moderately severe earthquakes are evaluated using the release categories in Table 3.15 and the parameters of the Evac-Reloc mode offsite emergency response in Table 4.1. The consequences and risks of accidents initiated by very severe regional earthquakes that could also affect the offsite conditions so as to seriously hamper evacuation or early relocation are also evaluated using the accident parameters in Table 3.15 and the parameters of the Late Reloc mode of offsite emergency response in Table 4.1

4.4 Consequences of Postulated Severe Accidents

Estimates of meteorology-averaged societal consequences, conditional upon releases of each category in Table 3.15, are shown in Table 4.2. For each release category, separate estimates are given using both of the offsite emergency response modes described in Table 4.1. These conditional mean values are useful in judging the relative severity of each release category, but cannot be used directly for risk assessment without also considering the probability of the release category causing the consequence. The risk (probability times consequence) contribution of each release category, and the total risk, including the probability-weighted consequences of all release categories, is discussed in Chapters 5 and 6.

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Table 4.2 Conditional Mean Values of Societal Consequences from Individual Release Categories for Two Alternative Offsite Emergency Response Modes

Consequence Category	Offsite Emergency Response	Conditional Mean Consequences For Release Categories									
		MiA	M1B	M2A	M2B	M4	M6	M6S†	M7	M7'*	M12
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	420	4.3	0.17	6.6	9.6	8.8	0	0	0	0
	Late-Reloc	**	***	55	13	260	22	†	1.1	1.4	***
2. Delayed cancer fatalities (including thyroid) (persons)	Evac-Reloc	2400	410	2800	4200	3000	2200	27	1400	1500	0.07
	Late-Reloc	**	***	3100	4600	3400	2500	†	1600	1700	***
3. Total person-rems	Evac-Reloc††										
	within -50 miles	5.0×10^{6}	1.8×10^{6}	5.5×10^{6}	4.5×10^{6}	6.5×10^{6}	4.5×10^{6}	1.4×10^{5}	3.3×10^{6}	3.3×10^{6}	-
	-350 miles	2.4×10^{7}	4.7×10^{6}	3×10^{7}	2.7×10^{7}	3.2×10^{7}	2.4×10^{7}	3×10^{5}	1.8×10^{7}	1.8×10^{7}	6×10^{2}
	Late-Reloc	**	***	3.4×10^{7}	3.1×10^{7}	3.6×10^{7}	2.7×10^{7}	†	2×10^{7}	2×10^{7}	***

†M6S is the case where electric power is recovered, resulting in the restoration of containment sprays leading to a hydrogen burn-induced containment failure. This release category is not applicable to the late reloc case.

††Person-rem dose for the Evac-Reloc case were calculated both for the areas within 50 miles and 350 miles of the plant.

*M7' is the case with iodine release fraction of 0.10 instead of 0.015.

**These release categories are initiated only by plant internal causes; therefore, the Late Reloc mode does not apply.

***This release category has a probability less than 10^{-9} per reactor-year to be initiated by severe earthquakes; therefore it is not analyzed with Late Reloc mode because the low probability will lead to an insignificant contribution to risk.

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5. SEVERE ACCIDENT RISK ESTIMATES

The conditional risk estimates listed in Table 4.2 are combined with the frequency of the containment release categories (Table 3.12) to calculate the risk contribution from each release category and to determine the total severe accident risk (Table 5.1). Estimates of total risk are shown in Table 5.2. By examining the overall risk estimates and the risk breakdowns contained in Table 5.1, conclusions are drawn about the overall safety of the Millstone Unit 3 plant and about those aspects of the plant where improvements need to be made.

5.1 Comparison of Severe Accident Risks With Other Societal Risks

To gain a perspective on the severity of the total risk estimates in Table 5.1, a comparison with related risks from non-nuclear sources is made.

Early fatality estimates can be compared with the risk of accidental death from other sources. The probability of accidental death for an individual in the United States is 5×10^{-4} per year. Based on this figure and the population distribution around the Millstone site, the expected number of accidental deaths within 1 mile of the plant's exclusion area boundary is 1 person per year. The corresponding expected number of early fatalities due to severe accidents at Millstone-3 within the same area is 6×10^{-5} persons per year,* based on the D&M seismic estimates or 2×10^{-4} based on the SHCP estimates. Clearly, the early fatality risk due to severe accidents at Millstone-3 is very small compared to other causes of accidental deaths.

The number of latent cancer fatalities can be compared with the annual cancer death rate from all causes. The probability of an individual dying from cancer in any given year is about 1.9×10^{-3}. Given the projected population within 50 miles of Millstone-3, one would expect about 6,300 cancer deaths per year. The estimated rate of 5.8×10^{-3} cancer deaths per reactor year within the same area as a result of postulated severe accidents at Millstone-3 based on Dames and Moore seismic estimates and 4.6×10^{-2} based on SHCP estimates is small by comparison.

5.2 Comparison of Risk With Other Nuclear Power Plants

The NRC requested a probabilistic safety study for Millstone-3, primarily because of concern about the high population density in the vicinity of the site. As in the cases of Zion, Indian Point and Limerick, the staff wanted to ascertain whether the Millstone 3 represents an undue portion of the risk of nuclear power. To help answer this question, we have compared the estimated risks for Millstone-3 with those of Zion and Indian Point and Limerick. These plants were chosen because they are located at high population sites, and because probabilistic safety studies for those plants, performed with methods comparable to those used for Millstone-3, have already been reviewed by the NRC staff. The risk estimates for all five plants are compared in Table 5.3.

*approximately 15% of all early fatalities that would occur following severe accidents can be expected within one mile of the plant.[17]

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Table 5.1 A Breakdown of Mean Annual Risk by Release Categories for Internal Events, Fires and Earthquakes (within 350 miles of the site boundary)*

Release Category	Internal	Fire	Seismic D & M	Seismic SHCP
i. Early Fatalities (Per Reactor Year)				
M1A	3.4(-4)	-	-	-
M1B	1.1(-5)	-	-	-
M4		-	2.2(-5)	9.6(-4)
M5/6	-	-	6.1(-6)	1.3(-4)
M6S	-	-	-	-
M7	-	-	3.1(-6)	6.2(-5)
ii. Latent Cancer Deaths (Per Reactor Year)				
M1A	1.9(-3)	-	-	-
M1B	1.0(-3)	-	-	-
M4	-	-	3.3(-4)	1.4(-2)
M5/6	-	-	7.6(-4)	1.6(-2)
M6S	1.2(-4)	-	-	-
M7	3.4(-3)	4.2(-3)	9.5(-3)	1.6(-1)
iii. Person-Rem (Per Reactor Year)				
M1A	19* [4]	-	-	-
M1B	12 [5]	-	-	-
M4	-	-	4 [1]	143 [26]
M5/6	-	-	8 [1]	165 [28]
M6S	1 [1]	-	-	-
M7	43 [8]	54 [10]	120 [21]	1998 [343]

*The numbers in brackets represent the mean annual person-rem dose within 50 miles of the plant. Otherwise the values are calculated to 350 miles.

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Table 5.2 Total Mean Annual Risk Estimates Based on D & M and SHCP Seismic Hazard Values

Risk Index (Per Reactor Yr)	Internal	Fire	Seismic		Total	
			D & M	SHCP	D & M	SHCP
Early Fatalities	3.5E-4*	-	3.1E-5	1.2E-3	3.8E-4	1.6E-3
Latent Fatalities	6.4E-3	4.2E-3	1.1E-2	1.9E-1	2.2E-2	2.0E-1
Public Dose** (Person-Rem)						
- within 50 miles	18	10	23	397	51	425
- within 350 miles	75	54	132	2306	261	2435

*3.5E-4 = 3.5×10^{-4} = .00035

**Public dose is quoted for the area within 50 miles of the plant and for the area within 350 miles of the plant.

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Table 5.3 Comparison of Millstone-3 Mean Risk Estimates with Zion, Indian Point and Limerick

Plant / Risk Index	Zion**	IP2†	IP3†	Limerick††	Millstone-3	
					D & M Seismic	SHCP Seismic
Early Fatalities	2(-4)*	1.5(-2)	3.8(-3)	5(-3)	3.8(-4)	1.6(-3)
Latent Cancer	1.8(-2)	2.1(-1)	1.1(-1)	8(-2)	2.2(-2)	2.0(-1)
Person-Rem	268	2600	1430	1000	261	2435

*2(-4) = 2×10^{-4} = 0.0004

†Janice E. Moore (NRC Counsel) to James P. Gleason, et al. (Administrative Judges), NRC Staff Witness Testimony (Witness, S. Acharya) on Commission Question 1 on Indian Point Units 1 and 2, January 24, 1983. Pages III.C.8 & 9, and Tables III.C.6 and 7.

††NUREG-0974; Final Environment Statement, Limerick Generating Station.

**NUREG/CR-3300, BNL-NUREG-51677 Vol. 2, W. T. Pratt, et al., "Review and Evaluation of the Zion Probabilistic Safety Study; Containment and Site Consequence Analysis, (Draft), July 1982.

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Millstone-3 risk estimates based on both the Dames and Moore and the SHCP seismic hazard functions are shown in Table 5.3. For purposes of comparison, the Dames and Moore numbers should be used, because that seismic calculation more closely approximates the methods used in estimating seismic hazard for Zion, Indian Point, and Limerick. The Millstone-3 risks based on the SHCP seismic method are much higher, but we assume that a reanalysis of the other plants with the LLNL seismic method could also yield higher risk estimates.

The Millstone-3 risk estimates are generally comparable to Zion and somewhat lower than Indian Point and Limerick, given the uncertainties in the data and methods used in calculating these risk estimates.

5.3 Dominant Contributors to Risk

The contributions of each release category to the three risk categories used in our comparisons (early fatalities, latent cancer fatalities and person-rem) are summarized in Table 5.1. Examination of these results reveals several interesting conclusions:

(a) For internal events, early fatality risks are dominated by the interfacing systems LOCA (release category M1A).

(b) Seismic events also contribute significantly to early fatality risks, particularly for the SHCP hazard function. These contributions are primarily from severe earthquakes which can directly fail containment (release category M4) and impede evacuation.

(c) Latent fatality and person-rem risks are dominated by late containment failure events (M7) due to overpressure failure or late hydrogen burns. A large fraction of these risks are estimated to result from seismically induced station blackout events. Those events will be examined in greater detail in Section 7 below.

5.4 Total Risk

The various risk indices are totaled and tabulated in Table 5.2. The SHCP-based early fatality estimates are about 4 times higher than the D&M based estimates, similarly the SHCP based latent fatalities and public dose are about an order of magnitude higher than their respective D&M estimates.

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6. UNCERTAINTIES

The use of point estimate values allows us to determine which plant systems, accident sequences and containment failure modes contribute most significantly to risk. However, in order to use those insights to make decisions about plant modifications and improvements, it is necessary to consider the uncertainties, biases and known errors in the point estimates. Uncertainties and errors can result from modeling assumptions, computational problems, omissions, and statistical fluctuations in random processes. The uncertainties arise in each step of the computational process; in the estimation of core melt frequencies, containment failure probabilities, radiological release fractions and offsite consequences. The purpose of this chapter is to explain and quantify the sources of these uncertainties (with the exception of core melt frequency), and to understand their potential impact on the uncertainty in overall risk.

6.1 Containment Failure Matrix

Because our actual experience with severe accidents at nuclear power plants is extremely limited, estimates of containment response must be based on phenomenological modeling and structural analysis. Nonetheless, for the large subatmospheric type of containment used at Millstone-3, there is a great deal of confidence that the response is well understood. The range of uncertainty is narrowed considerably by three factors, namely: (1) for the majority of accident sequences, operation of the recirculation sprays dramatically reduces containment failure probability and radioactivity available for release (2) there is little likelihood that the major penetrations to this type of containment will not be isolated at the start of an accident, and (3) the probability of early overpressure failure due to a "steam spike" has been shown by analysis to be minimal.[8]

Sequences without spray operation are dominated by station blackouts. The risk-dominant mode of containment failure for this sequence is a hydrogen burn failure following deinerting by natural condensation. Our analysis predicted a very low probability of containment failure by this mode, for two reasons: (1) containment sprays are likely to be recovered before the containment is deinerted by natural condensation, and (2) the resulting hydrogen burn is likely to be thermally inefficient due to the high steam concentrations. Under these assumptions, station blackout contributes about 9 person-rem per reactor year, approximately 12% of the total risk from internal events. If, however, all station blackouts lasting more than six hours after core melt are assumed to result in condensation deinerting and containment failure, the risks from station blackout would be 240 person rem per reactor year, and the overall risk from internal events would be increased by a factor of four.

A potentially significant source of uncertainty, and one which is difficult to quantify, is the possibility of early containment failure due to direct heating of the atmosphere by core debris dispersed from the reactor cavity. Recent experiments at Sandia National Laboratory indicate that failure of the reactor vessel lower head when the primary system is at high pressure

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will result in large portions of the core debris being lifted as aerosols into the upper regions of containment. The small corium particles would directly heat the atmosphere and chemically react with oxygen and steam. Calculations performed for the NRC's Containment Loads Working Group show that early containment failure is possible if a large fraction of the core can be suspended as aerosols for sufficient time.[13]

If direct heating induced containment failure were to occur for all high pressure sequences and the release fractions were characteristic of an M2B release category (α-failure mode), the estimates of early and latent fatalities together with the public dose would increase significantly for internally initiated events.

The scientific community is divided on whether such conditions can be attained, given all the obstructions in the lower part of containment (the highly confined reactor cavity at Millstone-3 would be a further impediment). There is also some evidence that high pressure failures of the reactor vessel lower head may be precluded. It has been argued that the high temperature steam produced during core boildown will produce breaks in the upper regions of the primary system which will relieve the pressure before the bottom head failure occurs.[14]

An additional source of uncertainty is the distribution function used by the applicant to describe the containment failure pressures (see Figure 3.2). There is concern that the low pressure end of the distribution gives undue weight to the possibility of failure at pressures below 100 psia. We have examined the dominant containment failure modes and found that variation of the failure pressure distribution within reasonable limits would not appreciably affect our risk estimates.

6.2 Radiological Source Terms

The amount, timing and energy of releases of radionuclides to the environment as a result of severe accidents are a source of considerable uncertainty in the estimation of risk. For the past decade, release fractions have been calculated with the CORRAL code, the method developed for the Reactor Safety Study.[10] Severe accident research conducted since the RSS has uncovered several phenomena which would tend to reduce the release fractions calculated by CORRAL. A systematic program to define a new methodology for source term estimation has been conducted in recent years by NRC's Accident Source Term Program Office (ASTPO). Part of this effort has been an uncertainty analysis (QUEST) conducted at Sandia National Laboratory.[15] During the period when the Millstone-3 review was conducted, the ASTPO methodology and QUEST uncertainty analysis were under review by a panel of the American Physical Society. Hence, the Millstone-3 review was conducted within the framework of the CORRAL methodology. Nonetheless, the ASTPO and QUEST results provide us with a useful source of information concerning source term uncertainties and their potential impact on overall risk.

The dominant release category contributing to latent cancer risk and person-rem is late failure without sprays (M7). For that scenario, the most effective mechanism for fission product removal is gravitational settling in

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the containment. The ASTPO codes calculate higher settling rates than the CORRAL code, mostly because the ASTPO codes predict enhanced agglomeration of aerosols, which enhances settling. However, the noble gases and organic iodine would not be significantly reduced by these mechanisms. Calculations of late containment failure for the Surry subatmospheric containment with the ASTPO methodology have yielded aerosol release fractions significantly less than the releases used in our risk calculations (Table 6.1). Furthermore, a preliminary QUEST analysis of this failure mode showed relatively little uncertainty. A range of cases was run by the QUEST program, and the results generally confirmed the low release rates for late failures.[15]

These results indicate that the uncertainty in the BNL/NRC releases are very high and all in the direction of much lower releases. The impact of this result will be discussed further in Section 7 below.

The principal source of early fatality risk is the interfacing system LOCA sequence (M1A). The release fractions assumed in arriving at that risk estimate took virtually no credit for deposition of fission products in the primary system or in the ECCS building, where the break was postulated to occur. To the extent that fission products are deposited in those two areas, and are not resuspended, the source term would be reduced, and with it the estimate of early fatalities. We have no firm basis for quantifying the extent of that uncertainty, except to say that it is almost certainly all in the direction of lower releases.

By contrast, the steam generator tube rupture (SGTR) source term (M1B), shows release fractions of a few percent for iodine, cesium, and tellurium. Such releases imply an order of magnitude reduction due to deposition in the primary system, steam generator secondary, steam piping and condenser. Actual radiological releases in SGTR events that lead to core melting are not well understood. Hence, it cannot be concluded that the M1B release fractions are conservative.

6.3 Consequence Analysis

Although the consequence model CRAC has been improved since the publication of the RSS, there are still large uncertainties in the results that are attributable to the consequence modeling, both in input to the model and in the model itself. A complete discussion of uncertainty in the consequence calculations is given in the Millstone-3 Final Environmental Study (NUREG-1064).

The relatively more important contributor to uncertainties in the results is atmospheric dispersion modeling for the radioactive plume, including the physical and chemical behavior of radionuclides in particulate form in the atmosphere. This uncertainty is due to differences between the modeling of the atmospheric dispersion of radioactivity in gaseous and particulate states in the CRAC code and the actual transport, diffusion and deposition that would occur during an accident (including the effects of precipitation). The phenomenon of plume rise because of heat that is associated with the atmospheric

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Table 6.1 Comparison of BNL/NRC Release Fractions for Millstone-3 with BMI-2104 Releases for Surry (Late Containment Failure)

Radionuclides	Release Fraction	
	BNL/NRC M7' Release for Millstone 3*	BMI-2104 (Vol. 5)† Surry[18]
Xe-Kr	9E-1	1E0
OI+1	1E-1*	2.8E-3
Cs-Rb	3E-1	3.9E-4
Te-Sb	3E-1	8.5E-2
Ba-Sr	3E-2	1.8E-2
Ru	2E-2	3.3E-6
La	4E-3	4.7E-4

*The M7' release differs from M7 insofar as the iodine release fraction is 1E-1 instead of 1.5E-2. This difference has minimal influence on severe accident consequences.

†These release fractions represent the amount of radiation released to the ground beneath the reactor due to a basemat meltthrough at 12 hours. They are indicative of the releases to the atmosphere if the failure mode had been overpressurization.

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release, effects of precipitation on the plume, and fallout of particulate matter from the plume all have considerable impact on both the magnitude of the early health consequences and the distance from the reactor to which these consequences would occur. It is our judgment that these factors can result in substantial overestimates or underestimates of both early and later effects (health and economic).

Other areas that have substantial but relatively less effect on uncertainty than the preceding items include the duration and energy of release, warning time, and in-plant radionuclide decay time. The assumed release duration, energy of release, and the warning and the in-plant radioactivity decay times may differ from those that would actually occur during a real accident.

It is believed that the uncertainty associated with the consequence calculation could cause substantial underestimates or overestimates of the early consequences and risks. The magnitude of the overall uncertainty is difficult to quantify, but it is believed to be between one and two orders of magnitude.

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7. ANALYSIS OF MITIGATIVE SYSTEMS

If design improvements for the purpose of mitigating severe accidents are to be required for Millstone-3, the decision will be based on the estimated cost of averted public dose. For the purpose of performing a preliminary screening analysis, the estimated person-rem per reactor year for each failure mode (Table 7.1) was assigned a monetary value based on $1000 per person-rem over an assumed 40-year lifetime. A complete regulatory cost/benefit analysis would not lead to qualitatively different results. Although the NRC method of cost benefit analysis treats only the risk within 50 miles of the plant, we also considered the risk within 350 miles because there are several large population areas beyond the 50-mile point and the societal risks could be substantially greater than indicated from only considering effects within 50 miles.

For various reasons, several of the release categories may be excluded from consideration. For instance, the dominant contributor to the M4 release category is direct containment failure due to seismic events with accelerations several times the safe shutdown earthquake (SSE). We do not believe we should contemplate a mitigative device which would be expected to operate under such severe circumstances.

The M6S release category was also disregarded because of its minimal impact on overall risk.

The M1A release category, interfacing system LOCA, leads to bypass of containment. Our analysis of radiological releases in the sequence gave little credit for the mitigative effects of features which already exist on the plant; i.e., fission product retention in the reactor vessel, primary system piping and ECCS building. Even under these conservative assumptions, the monetized value of person-rem within 50 miles (Table 7.1) is $160 thousand* ($800 thousand for person-rem within 350 miles). Taking credit for existing mitigative features, it is difficult to justify consideration of design changes based on the level of risk.

Finally, mitigation devices for the M1B release category (steam generator tube ruptures) are also excluded from consideration. Core melt accidents in this category are dominated by operator errors. The staff has decided to pursue a reduction in core melt frequency through operator training.

Having eliminated several release categories from consideration, we conducted detailed screening cost/benefit analyses for two: hydrogen burn failure at intermediate time (M6) and late containment failure (M7).

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*4 person-rem per reactor year x 40 years x $1,000 per person-rem.

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Table 7.1 Mean Annual Public Exposure Risk Estimates from Various Containment Failure Modes for Internal Events, Fires and Seismic Events

Failure* Mode	Mean Annual Public Exposure (Person-rem per Reactor Year)			
	Internal Events	Fires	Seismic	
			D & M	SHCP
M1A	19† [4]	-	-	-
M1B	12 [5]	-	-	-
M4	-	-	4 [1]	143 [26]
M5/6	-	-	8 [1]	165 [28]
M6S	1 [1]	-	-	-
M7	43 [8]	54 [10]	120 [21]	1998 [343]

*M1A: Interfacing Systems LOCA
M1B: Steam Generator Tube Rupture
M4: Seismic Induced Crane Wall Failure
M5/6: Intermediate Failure due to Hydrogen Burn
M6S: Hydrogen Burn Following Recovery of Sprays in a Station Blackout
M7: Late Failure Due to Overpressurization or Hydrogen Burn

†Numbers of brackets represent person-rem dose integrated over the area within 50 miles of the plant. Un-bracketed estimates are integrated to 350 miles from the plant.

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7.1 Hydrogen Burn Failure at Intermediate Time (M6)

The M6 release category results from severe seismic events which lead to large LOCA's in coincidence with station blackout. Detailed containment thermal-hydraulic calculations by the applicant showed that hydrogen burns would be possible at about four hours after reactor trips. They assigned a 62% likelihood to hydrogen burn failure at that point.

For the purpose of estimating risk, the staff and BNL accepted the applicant's analysis of this sequence. However, for the purpose of evaluating mitigative actions, we have reconsidered the accident phenomenology. Our estimate of hydrogen generation during core meltdown and after vessel failure lead us to conclude that a hydrogen burn at four hours would be very unlikely to fail containment, and the actual failure would occur much later due to overpressurization. To fail containment, a hydrogen burn would have to occur more thar six hours after vessel failure, and probably much later when the hydrogen produced in-vessel and the continuing hydrogen production ex-vessel due to core-concrete interactions would lead to high hydrogen concentrations in the containment.

In either case, we conclude that the core melt sequence currently assigned to the M6 release category would actually lead to late containment failure instead. Consequently, for the purpose of evaluating mitigative systems, we have grouped the M6 and M7 release categories under the general heading of late containment failure.

7.2 Late Containment Failure (M6 and M7)

Late containment failures result from both seismic and non-seismic initiators. These two categories are treated separately in the discussion below because the seismic sequences have greater uncertainty and the cost of seismically qualified design changes are expected to be higher.

7.2.1 Non-Seismic Initiators

The person-rem risk from non-seismic initiators comes from a variety of sequences, including station blackouts, fires, and LOCA's with failure of recirculation spray. Late failures result from both overpressurization and hydrogen burns. There are two types of design improvements which would be capable of mitigating all of these scenarios. A filtered-vent would prevent overpressure failure by relieving the pressure in containment and prevent hydrogen burn failure by removing considerable quantities of oxygen.

A dedicated AC-independent spray pump system would significantly reduce the radiological releases due to the late failure. Furthermore, the spray would prevent overpressure failures by condensing steam. The effect of sprays on hydrogen burn failure probability is ambiguous. On the one hand, the resulting steam condensation would enhance the probability and magnitude of hydrogen burns, but on the other hand, spray operation could lead to early hydrogen burns before sufficient hydrogen is available to fail containment. Nonetheless, the reduction in source term would always lead to reduced levels of risk.

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Accurate cost estimates for these two systems are difficult to obtain. Reference [16] presents cost estimates for both the filtered vented system and a direct diesel driven pump spray system. The estimates are based on somewhat different applications, and are for the BWR Mark II containment, but the costs should be indicative. The cost of each system was estimated at about 4 million dollars. The study described in Reference [16] attempted to evaluate the minimum costs for mitigative features. The equipments selected were normal industrial stock. The cost to qualify the equipment as safety related was intentionally disregarded. The costs included only installation, with no accounting for maintenance and surveillance. Finally, it was assumed that the entire installation would be accomplished during refueling. Hence, there was no allowance for the cost of replacement power. Consequently, we assume that these cost estimates represent a lower limit.

We have no sound basis for estimating an upper limit. Qualifying equipment as safety related can multiply the cost by as much as a factor of ten. Again the staff does not intend to require such qualification. If installation extended the refueling time by one week the replacement power cost would be about $5 million. Maintenance and surveillance costs over the 30-year lifetime of the plant would also be substantial.

For perspective the FILTRA vent installed on the Barsbeck containment in Sweden cost about $20 million (Figure 7.1a).

The person-rem risk for internal events and fires (Table 7.1) ranges from 18 person-rem per reactor year within 50 miles of the plant to 97 person-rem per reactor year within 350 miles. The monetized values of these two estimates are $720 thousand* and $3.9 million, respectively. This range of values is plotted on Figure 7.1a together with the cost estimates discussed above. As seen in Figure 7.1a, the monetized risk due to non-seismic late containment failures is generally lower than the range of estimated costs for mitigation devices.

Furthermore, our estimates of conditional consequences for late failures are based on source term estimates derived with WASH-1400 methodology. Minimal credit for aerosol agglomeration and gravitational settling is included in those estimates. The revised NRC source term methodology[19] claims considerable credit for that mechanism. The new methodology also includes mechanisms which would tend to increase the releases of refractory fission products. For early containment failure scenarios, it is not clear what the net effect of these competing mechanisms would be. However, for the late containment failures considered here, the revised methodology would unambiguously predict substantially lower source terms than we have assumed in our risk estimate.

We conclude that the cost effectiveness of a filtered vent or AC-independent spray system would be even less favorable than Figure 7.1 indicates if new source term information were used, and we would not recommend either system for mitigation of non-seismic events. Less expensive systems that might be cost effective will be discussed below in connection with seismic events.

*18 person-rem per reactor year x 40 years x $1000 per person-rem.

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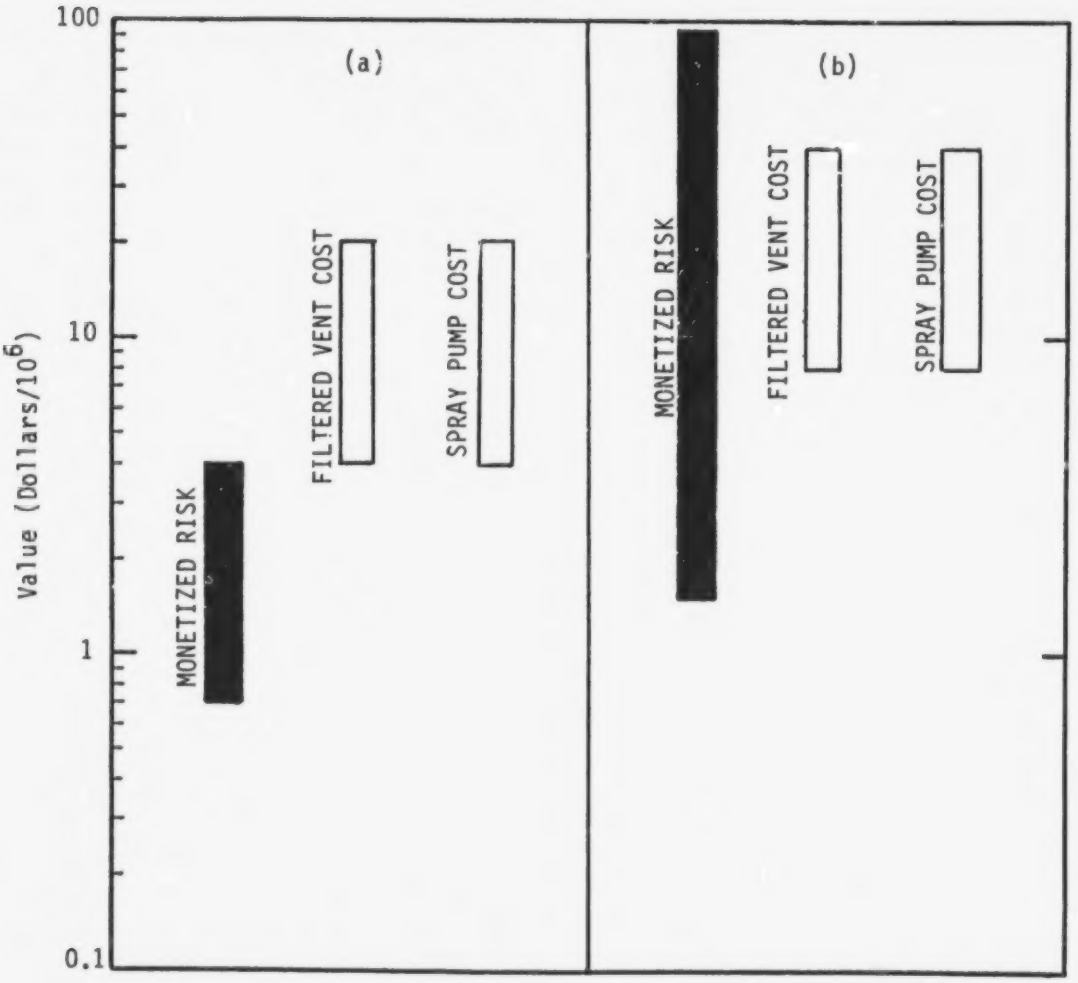

Figure 7.1 Cost and Benefit Estimates and Uncertainty Ranges of Mitigative Features Which Would Provide Protection for (a) Non-Seismic Overpressure Failure and Hydrogen Burns, and (b) Seismically Induced Overpressure Failure.

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7.2.2 Seismic Initiators

Seismically induced late failures are dominated by station blackout sequences. Late failure would result from hydrogen burns, or failing that, from late overpressurization. Mitigative devices to prevent or reduce hydrogen burns alone would be of limited usefulness because the containment would eventually fail due to overpressure.

The prime candidates for mitigation of seismic events are the same examined for non-seismic sequences: filtered venting and AC-independent containment spray. To be effective in mitigating seismic events, the design improvements would have to have a seismic capacity of 1.5 g or greater. We estimate that this extra requirement would roughly double the cost (Figure 7.1b).

The risk reduction achievable would be the sum of seismic and non-seismic events for both the M6 and M7 release categories (Table 7.1). These estimates range from a low value of 40 person-rem per reactor year based on the PSS hazard curve and including only the risk within 50 miles of the plant, to 2260 person-rem per reactor year for the SHCP hazard curve including risk within 350 miles. The monetized values of those risk estimates are $1.6 million and $90 million, respectively, are plotted in Figure 7.1b.

The comparison of costs and benefits in Figure 7.1b indicates that either of the two mitigative fixes might be cost effective. This result prompted us to more closely examine the estimates of both cost and risk.

For a number of reasons, we have concluded that the risk estimates shown in Figure 7.1b are greatly overestimated. First, a reduction of about a factor of two was obtained by reexamining the seismic fragility of some key components. One important failure point, the diesel generator room footings, were found to have been evaluated too conservatively. Another weakness, the bolts on the diesel generator oil cooler, could be eliminated with an inexpensive design change which the staff intends to recommend.

More importantly, for reasons discussed in Sections 6.2 and 7.2.1, we are confident that the radiological releases have been overestimated by an order of magnitude or more. This conclusion is based on results of the revised NRC source term methodology.

Taking these factors into account, we estimate that mitigative measures in the vicinity of $1 million in cost would be justified. It is not reasonable to expect that a seismically qualified filtered vent could be installed for that price. It is possible, however, that some type of AC-independent spray system could be installed. For instance, a fire truck could be expected to survive a severe seismic event and could serve as part of a mobile, manually-operated, AC-independent spray system, taking suction from Long Island Sound.

The results of our cost/benefit screening analysis indicate that such a system should be examined in more detail. A more careful analysis of risk should be performed including consideration of any potential for increased

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risk due to the system. More accurate cost figures should be obtained for a variety of options including mobile and stationary pumps, both automatic and manually operated. This proposed change will be discussed in the NRC staff report on the risk of the Millstone-3 plant.

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8. SUMMARY

The purpose of this report is to describe the technical review of the Millstone-3 probabilistic risk study (MPSS-3) and to present our estimates of containment performance, radiological source terms, offsite consequences and risk.

The containment response to severe accidents is judged to be an important factor in reducing risk. There is negligible probability of prompt containment failure or failure to isolate. Failure during the first four hours after core melt is also unlikely. Late overpressure failures are also unlikely, except for seismically initiated sequences. Most core melt accidents would be effectively mitigated by containment spray operation.

Offsite consequences of core melt accidents with containment failure are relatively high because of the high population surrounding the site.

The estimated overall early and latent fatality risk from severe accidents at Millstone-3 is extremely low compared with non-nuclear sources of risk. Severe accident fatality risk for Millstone-3 are comparable to the risk from other plants at high population sites. Seismically induced accidents dominate the latent fatality risk of Millstone-3.

Cost-benefit screening analyses for various potential mitigative features have been performed based on a range of assumptions about risk and cost estimates. In general, the mitigative features are not cost-effective. Our preliminary assessment indicates that an inexpensive containment spray pumping system should be carefully evaluated as a potential mitigative feature for late containment failures.

The final determination of the desirability of mitigative features involves factors which are forthcoming beyond the scope of this report. That determination will be discussed in the forthcoming NRC staff report on the regulatory implications of the risk analysis for the Millstone-3 plant.

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A technical review and evaluation of the Millstone Unit 3 Probabilistic Safety Study has been performed. It was determined that; (1) long-term damage indices (latent fatalities, person-rem, etc.) are dominated by late failure of the containment, (2) short-term damage indices (early fatalities, etc.) are dominated by bypass sequences for internally initiated events, while severe seismic sequences can also contribute significantly to early damage indices. These overall estimates of severe accident risk are extremely low compared with other societal sources of risk. Furthermore, the risks for Millstone-3 are comparable to risks from other nuclear plants at high population sites. Seismically induced accidents dominate the severe accident risks at Millstone-3. Potential mitigative features were shown not to be cost-effective for internal events. Value-impact analyses for seismic events showed that a manually actuated containment spray system might be cost-effective.

17. KEY WORDS AND DOCUMENT ANALYSIS

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